3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.15 At least one of the following Overpressure Protection Systems shall be OPERABLE, with the accumulators isolated:

a. The Normal Residual Heat Removal System (RNS) suction relief valve, or

b. The RCS depressurized and an RCS vent of ≥ [5.4] square inches.

----------NOTE----------
When the RCS temperature is ≥ 200°F, a reactor coolant pump (RCP) may not be started if the pressurizer level is ≥ 92%.

APPLICABILITY: MODE 4 when any cold leg temperature is ≤ 275°F,
MODE 5,
MODE 6 when the reactor vessel head is on.

----------NOTE----------
Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. An accumulator not isolated when the accumulator pressure is ≥ to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	A.1 Isolate affected accumulator.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Increase RCS cold leg temperature to a level acceptable for the existing accumulator pressure allowed in the PTLR.	12 hours
	OR	
	B.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours
C. The RNS suction relief valve inoperable.	C.1 Restore the RNS suction relief valve to OPERABLE status.	12 hours
	OR	
	C.2 Depressurize RCS and establish RCS vent of ≥ [5.4] square inches.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Verify each accumulator is isolated.	12 hours
SR 3.4.15.2	Verify both RNS suction isolation valves in one RNS suction flow path are open.	12 hours
SR 3.4.15.3	----------NOTE---------- Only required to be performed when complying with LCO 3.4.15.b. ---------- Verify RCS vent ≥ [5.4] square inches is open.	12 hours for unlocked-open vent AND 31 days for locked-open vent
SR 3.4.15.4	Verify the lift setting of the RNS suction relief valve.	In accordance with the Inservice Testing Program

RAI Number 630.023

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System limits RCS pressure at low temperatures so that the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the limits which set the maximum allowable setpoints for the Normal Residual Heat Removal System (RNS) suction relief valve. LCO 3.4.3 provides the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a maximum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires isolating the accumulators. The pressure relief capacity requires the RNS suction relief valve or a depressurized RCS and an RCS vent of sufficient size. The RNS suction relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

(continued)

RAI Number 630.023-6

BASES

BACKGROUND (continued)

RNS Suction Relief Valve Requirements

During the LTOP MODES, the RNS system is operated for decay heat removal. Therefore, the RNS suction isolation valves are open in the piping from the RCS hot legs to the inlet of the RNS system. While these valves are open, the RNS suction relief valve is exposed to the RCS and able to relieve pressure transients in the RCS.

The RNS suction relief valve is a spring loaded, water relief valve with a pressure tolerance and an accumulation limit established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.

The RNS suction isolation valves must be open to make the RNS suction relief valves OPERABLE for RCS overpressure mitigation.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the flow capacity requirement, it may require removing one or more pressurizer safety valves or manually opening one or more Automatic Depressurization System (ADS) valves. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with the RCS temperature above 275°F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. When the RNS is aligned and open to the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS, overpressure protection is provided by the RNS suction relief valve, or a depressurized RCS and a sufficiently sized open RCS vent.

The actual temperature at which the pressure in the P/T limit curve falls below the suction relief setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the RNS suction relief valve, or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients. The events listed below were used in the analysis to size the RNS suction relief valve. Therefore, any events with a mass or heat input greater than the listed events cannot be accommodated and must be prevented.

Mass Input

a. Makeup water flow rate to the RCS assuming both CVS makeup pumps are in operation and letdown is isolated.

Heat Input

a. Restart of one reactor coolant pump (RCP) with water in the steam generator secondary side 50°F hotter than the primary side water, and the RCS water solid.

RNS Suction Relief Valve Performance

Since the RNS suction relief valve does not have a variable P/T lift setpoint, the analysis must show that with chosen setpoint, the relief valve will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the minimum of either the P/T limit curve or 110 percent of the design pressure of the normal residual heat removal system. The current analysis shows that up to a temperature of 100°F, the mass input transient is limiting, and above this temperature the heat input transient is limiting.

(continued)

RAI Number 630.023

BASES

APPLICABLE SAFETY ANALYSES (continued)

To prevent the possibility of a heat input transient, and thereby limit the required flow rate of the RNS suction relief valve, an administrative requirement has been imposed that does not allow an RCP to be started with the pressurizer water level above 92% and the RCS temperature above 200°F. Under these imposed conditions, the transient created by the startup of an RCP when the RCS temperature is above 200°F can be accommodated without additional pressure relief.

RCS Vent Performance

With the RCS depressurized, a vent size of [5.4] square inches is capable of mitigating a limiting overpressure transient. The area of the vent is equivalent to the area of the inlet pipe to the RNS suction relief valve so the capacity of the vent is greater than the flow possible with either the mass or heat input transient, while maintaining the RCS pressure less than the minimum of either the maximum pressure on the P/T limit curve or 110 percent of the design pressure of the normal residual heat removal system.

The required vent area may be obtained by opening one ADS flow path.

The RCS vent size will be reevaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the maximum coolant input and minimum pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires all accumulator discharge isolation valves closed and immobilized, when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed in the PTLR.

(continued)

RAI Number 630.023

BASES

LCO (continued)

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. One OPERABLE RNS suction relief valve; or

An RNS suction relief valve is OPERABLE for LTOP when both RNS suction isolation valves in one flow path are open, its setpoint is within limits, and testing has proven its ability to open at this setpoint.

b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of ≥ [5.4] square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 when any cold leg temperature is below 275°F, MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above 275°F. In MODE 6, the reactor vessel head is off, and overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.7, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 with the RNS isolated or RCS temperature ≥ 275°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure with little or no time for operator action to mitigate the event.

The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves.

This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

(continued)

RAI Number 630.023

BASES (continued)

ACTIONS

A.1, B.1, and B.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action B.1 and Required Action B.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > 275°F, the accumulator pressure cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

C.1 and C.2

If the RNS suction relief valve is inoperable and the RCS is not depressurized, there is a potential to overpressurize the RCS and exceed the limits allowed in LCO 3.4.3. The suction relief valve is considered inoperable if the RNS isolation valves have isolated the RNS from the RCS in such a way that the suction relief valve cannot perform its intended safety function, or if the valve itself will not operate to perform its intended safety function.

Under these conditions, Required Actions C.1 or C.2 provide two options, either of which must be accomplished in 12 hours. If the RNS suction relief valve cannot be restored to operable status, the RCS must be depressurized and vented with a RCS vent which provides a flow area sufficient to mitigate any of the design low temperature overpressure events.

The 12 hour Completion Time represents a reasonable time to repair the relief valve, open the RNS isolation valves or otherwise restore the system to OPERABLE status, or depressurize and vent the RCS, without imposing a lengthy period when the LTOP system is not able to mitigate a low temperature overpressure event.

(continued)

RAI Number 630.023

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, the accumulator discharge isolation valves are verified closed and locked out. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the main control room to verify the required status of the equipment.

SR 3.4.15.2

The RNS suction relief valve shall be demonstrated OPERABLE by verifying two RNS suction isolation valves in one flow path are open. This Surveillance is only performed if the RNS suction relief valve is being used to satisfy this LCO.

The RNS suction isolation valves are verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RNS suction isolation valves remain open.

SR 3.4.15.3

The RCS vent of ≥ [5.4] square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is not locked (valves that are sealed or secured in the open position are considered "locked" in this context) or

b. Once every 31 days for other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position or a removed pressurizer safety valve or open manway also fits this category).

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.15b.

SR 3.4.15.4

The RNS suction relief valve shall be demonstrated OPERABLE by verifying that two RNS suction isolation valves in one flow path are open and by testing it in accordance with the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.4 (continued)

Inservice Testing Program. (Refer to SR 3.4.15.2 for the RNS suction isolation valve Surveillance.) This Surveillance is only required to be performed if the RNS suction relief valve is being used to meet this LCO. The ASME Code, Section XI (Ref. 5), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix G, "Fracture Toughness Requirements."

2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operation."

3. ASME Boiler and Pressure Vessel Code, Section III.

4. Section 5.2.2, "Overpressure Protection."

5. ASME, Boiler and Pressure Vessel Code, Section XI.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.024

Question:

(Section 15.4.4; Section 16.1, TS LCO 3.4.4 Note 1 & Bases) The accident analysis for startup of an inactive reactor coolant pump (RCP) at an incorrect temperature relies on Note 1 of TS 3.4.4 (no starting of a RCP with the reactor trip breakers [RTBs] closed) to ensure that initial core power is approximately zero megawatts (thermal) [MWt] and that no increase in core power will occur. However, the Bases for this note states that "[t]his requirement prevents circulation of cold and/or unborated water from an inactive loop into the core, which could cause a reactivity excursion." According to the Bases for Required Actions A.1 and B.2, starting an RCP with the reactor subcritical, with the RTBs closed, and possibly one or more withdrawn control rods, or critical below P-10, has not been analyzed, and is apparently deemed not credible because of TS LCO 3.4.4 Note 1. Please consider revising the Bases discussion of Note 1 to be more consistent with the Chapter 15 discussion. In addition:

A. Why was this restriction not explicitly included in the AP1000 TSs 3.4.4 and 3.4.5? The AP1000 TSs contain the restriction of Note 1, but only for entry into Mode 2 from Mode 3, by virtue of LCO 3.0.4 and SR 3.0.4 in combination with LCO 3.4.4.

B. The Notes in Conditions A and B of the AP1000 TS 3.4.4 accomplish the same preventive action as Note 1. That is, they do not allow exiting the Condition by restarting a RCP with the RTBs still closed. Consider whether Note 1 is needed.

C. Why is there no interlock that prevents starting an RCP with the RTBs closed?

D. Why is Note 1 a sufficient restriction to ensure this event could never happen?

E. When is it necessary to close the RTBs and withdraw a control rod in Modes 4 and 5?

Westinghouse Response:

The discussion for Note 1 in the LCO section of the Bases for TS 3.4.4 has been revised to be consistent with the discussion in DCD Section 15.4.4.

A. The AP1000 RCP starting restriction is explicitly included in AP1000 TS 3.4.4. There is no TS 3.4.5 for AP1000. The AP1000 TS 3.4.4 is a combination of the AP600 TSs 3.4.4 and 3.4.5.

Due to the increased size of the AP1000 RCPs, a variable speed controller is needed during startup and operation of the RCP during cold plant conditions below 450°F. The variable speed RCP controller was not needed for the AP600 RCPs. TS 3.4.4 requires that the RCPs be operating at full speed operation with the variable speed controller bypassed in MODES 1 and 2, and whenever the reactor trip breakers are closed in MODES 3, 4, and 5.

The restrictions in Conditions A and B were added to TS 3.4.4 during the development of the AP1000 TSs, based on the addition of the nonsafety-related RCP variable speed controller. The startup restrictions were specifically added to minimize the potential for unanalyzed controller malfunctions during RCP startup. The restrictions eliminated the need to perform comprehensive evaluations of credible controller failure mechanisms and associated analyses during the development of the AP1000 DCD and TSs. The requirements of Conditions A and B preclude the potential for unanalyzed RCP startup events.

B. Note 1 was intentionally added for explicit clarification to the operators, recognizing that the same requirements are contained in the Required Actions for Conditions A and B. The note is very concise, explicit, and highly visible to the operators, which can be especially important in preventing the need to review or interpret Required Actions versus plant conditions when considering plant response and recovery following the loss of an RCP. The requirements of Note 1 apply for any MODE where a RCP is started, which is more restrictive than the requirements of LCO 3.0.4 and SR 3.0.4.

C. The TS requirements of Conditions A and B are more than sufficient to prevent starting an RCP with the RTBs closed, without the need to add redundant electrical interlocks.

An interlock is not required since the restriction is based on being in an unanalyzed condition for AP1000, rather than being in a condition that is expected to present a significant challenge to plant safety. Startup of an idle RCP is allowable in current operating plants in MODES 1 to 5 with no restrictions on reactor power or control rod status. If a RCP trips during power operation, it can be restarted in MODE 1 with the reactor critical, and without any requirement to place the plant in MODE 2 or 3 for the restart. In addition, startup of an idle RCP was analyzed for AP600 and found to be acceptable. The safety impact on AP1000 is expected to be relatively the same as AP600 and currently licensed plants. However, startup of an idle RCP is not specifically required to be analyzed for AP1000 since the restrictions in TS 3.4.4 preclude this event.

D. Note 1 is a sufficient restriction for precluding RCP startup with the reactor trip breakers closed since the TS requirements, including Note 1, establish the basis for the plant operational procedures that must be followed by the operators, as well as the licensing basis requirement for RCP startup and operation. TSs establish one of the most fundamental and significant sets of operational requirements and restrictions that must be followed by the plant operators, and TSs are completely sufficient in precluding this event from occurring.

E. As part of the normal plant startup procedure, the reactor trip breakers are typically closed in MODE 5, after a steam bubble is established in the pressurizer and the RCPs are started to begin the plant heatup.

The RCS must be at approximately 400 psig to start any RCPs and at least one RCP is normally required to be operating whenever RCS temperature is above 160°F. After starting the RCPs, the trip breakers are then closed so that the shutdown rods can be fully withdrawn, which typically is done shortly after the RCPs are started, but in MODE 5 before the RCS reaches 200°F during the plant heatup.

The control rods are not withdrawn in MODE 4 or 5. The control rods are only withdrawn in MODE 3 as part of the reactor startup and transition to MODE 2, after RCS boron has been reduced to the critical boron concentration. Control rod withdrawal to criticality is only permitted with all RCPs operating and with the RCS at normal operating temperature and pressure, in accordance with the requirements of TSs 3.4.1, 3.4.2 and 3.4.4.

Design Control Document (DCD) Revision:

TS B 3.4.4, LCO Section will be revised as follows:

"Note 1 prohibits startup of a RCP when the reactor trip breakers are closed. This requirement prevents **startup of a RCP and the resulting** circulation of cold and/or unborated water from an inactive loop into the core, ~~which could cause~~**precluding** ~~a~~ reactivity excursion~~.~~**events which are unanalyzed.**

PRA Revision:

None

BASES

LCO (continued)

flow in the core to avoid exceeding the DNBR limit. Bypass of the RCP variable speed control ensures that the pumps are operating at full flow.

With the RTBs in the open position, the PLS is not capable of rod withdrawal; therefore only a minimum RCS flow of 10,000 gpm is necessary to ensure removal of decay heat from the core in accordance with LCO 3.4.9, Minimum RCS Flow.

Note 1 prohibits startup of a RCP when the reactor trip breakers are closed. This requirement prevents startup of a RCP and the resulting circulation of cold and/or unborated water from an inactive loop into the core, precluding reactivity excursion events which are unanalyzed.

Note 2 requires that the secondary side water temperature of each SG be ≤ [50]°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ 275°F. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 3 permits all RCPS to be de-energized in MODE 3, 4, or 5 for ≤ 1 hour per 8 hour period. The purpose of the NOTE is to permit tests that are designed to validate various accident analysis values. One of these tests is for the validation of the pump coastdown curve, used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve may need to be revalidated by conducting the test again.

Another test performed during the startup testing program is the validation of the rod drop times during cold conditions, both with and without flow.

The no-flow tests may be performed in MODE 3, 4, or 5, and require that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should only be performed once, unless the flow characteristics of the RCS are changed. The 1 hour time

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.025

Question:

(Section 16.1, Bases for TS 3.4.17 Required Action C.2) The Completion Time of Required Action C.2 seems to allow for cooling down to the temperature (assumed to be 275°F) at which the residual heat removal system (RNS) may be placed in operation after entering Mode 4. The Bases should explain this more explicitly because the usual time to reach Mode 4 (420°F) on a shutdown action is 12 hours.

Westinghouse Response:

Action time for MODE 4 entry will be changed, as indicated below, to 12 hours.

In addition, the wording in the Bases for TS 3.4.17 Action B.1 was in error and has been corrected, as indicated below.

Design Control Document (DCD) Revision:

From DCD Chapter 16, TS 3.4.17, pg. 3.4-32:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Reactor Vessel Head Vent (RVHV)

LCO 3.4.17 The Reactor Vessel Head Vent shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with the RCS not being cooled by the RNS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One flow path inoperable.	A.1 Restore flow path to OPERABLE status.	72 hours
B. Two flow paths inoperable.	B.1 Restore at least one flow path to OPERABLE status.	6 hours
C. Required Action and associated Completion Time not met. OR Requirements of LCO not met for reasons other than Conditions A or B.	C.1 Be in MODE 3. AND C.2 Be in MODE 4, with the RCS cooling provided by the RNS.	6 hours ~~24~~12 hours

From DCD Chapter 16, TS B 3.4.17, pg. B 3.4-72:

ACTIONS A.1

If one or two RVHV valves in a single flow path are determined to be inoperable, the flow path is inoperable. The remaining OPERABLE RVHV flow path is adequate to perform the required safety function. A Completion Time of 72 hours is acceptable since the OPERABLE RVHV paths can mitigate DBAs without a single failure.

B.1

If **both**~~two~~ flow paths are determined to be inoperable, the RVHV is degraded such that the ~~remaining~~ system ~~capacity may not be~~**is not**~~adequate~~ **available** for some DBA non-LOCA analysis **for which may be required.** A Completion Time of 6 hours is permitted to restore at least one flow path. This Completion Time is acceptable considering that the realistic analysis of these non-LOCA events do not result in pressurizer overfill.

From DCD Chapter 16, TS B 3.4.17, pg. B 3.4-73:

ACTIONS (continued) C.1 and C.2

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.17 are not met for reasons other than Conditions A or B, the plant must be brought to MODE 4 where the probability and consequences of an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within ~~24~~**12** hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner, without challenging plant systems.

PRA Revision:

None

RAI Number 630.025

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Reactor Vessel Head Vent (RVHV)

LCO 3.4.17 The Reactor Vessel Head Vent shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with the RCS not being cooled by the RNS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One flow path inoperable.	A.1 Restore flow path to OPERABLE status.	72 hours
B. Two flow paths inoperable.	B.1 Restore at least one flow path to OPERABLE status.	6 hours
C. Required Action and associated Completion Time not met. OR Requirements of LCO not met for reasons other than Conditions A or B.	C.1 Be in MODE 3. AND C.2 Be in MODE 4, with the RCS cooling provided by the RNS.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify that each RVHV valve is OPERABLE by stroking it open.	In accordance with the Inservice Testing Program

BASES (continued)

LCO

The requirement that all four RVHV valves be OPERABLE ensures that upon actuation, the RVHV can reduce the pressurizer water level as assumed in the DBA safety analyses.

For the RVHV to be considered OPERABLE, all four valves must be closed and OPERABLE (capable of opening from the main control room).

APPLICABILITY

In MODES 1, 2, 3, and 4 with the RCS not being cooled by the RNS, the RVHV must be OPERABLE to mitigate the potential consequences of any event which causes an increase in the pressurizer water level that could otherwise result in overfilling of the pressurizer.

In MODE 4, with the RCS being cooled by the RNS, and in MODES 5 and 6, operation of the CMTs or CVS will not result in a pressurizer overfill event.

ACTIONS

A.1

If one or two RVHV valves in a single flow path are determined to be inoperable, the flow path is inoperable. The remaining OPERABLE RVHV flow path is adequate to perform the required safety function. A Completion Time of 72 hours is acceptable since the OPERABLE RVHV paths can mitigate DBAs without a single failure.

B.1

If both flow paths are determined to be inoperable, the RVHV is degraded such that the system is not available for some DBA non-LOCA analysis for which may be required. A Completion Time of 6 hours is permitted to restore at least one flow path. This Completion Time is acceptable considering that the realistic analysis of these non-LOCA events do not result in pressurizer overfill.

(continued)

RAI Number 630.025

BASES

ACTIONS (continued)

C.1 and C.2

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.17 are not met for reasons other than Conditions A or B, the plant must be brought to MODE 4 where the probability and consequences of an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner, without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

The dedicated component level remote manual valve switches in the main control room shall be used to stroke each RVHV valve to demonstrate OPERABILITY of the controls.

This Surveillance requires verification that each RVHV valve strokes to its fully open position. The Surveillance Frequency for demonstrating valve OPERABILITY references the Inservice Testing Program.

REFERENCES

1. Section 15.5, "Increase in Reactor Coolant System Inventory."

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.026

Question:

(Sections 15.5.1 and 15.5.2) The analyses of incidents [or transients] that increase reactor coolant inventory, based on the descriptions provided, assume pressurizer spray operation and automatic rod control. If these incidents present a challenge to the integrity of a fission product barrier, and the spray and rod control are part of the primary success path which functions to mitigate such design-basis transients, then it seems these two functions satisfy 10 CFR 50.36(c)(2)(ii)(C) Criterion 3. Please explain the rationale for not explicitly addressing the spray and rod control in a TS LCO.

Westinghouse Response:

As reported in Section 15.0.8, except for the cases reported in Table 15.0-8, non-safety-related systems are not used for mitigating the consequences of an accident. Control systems are assumed to function only if their operation results in more severe analytical results. This methodology is also followed for the accidents described in DCD sections 15.5.1 "Inadvertent Operation of the Core Makeup Tanks During Power Operation" and 15.5.2 "Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory."

For the Inadvertent Operation of the Core Makeup Tanks During Power Operation, as reported on page 15.5-3, both rod control and pressurizer spray are conservatively assumed to operate. Based on sensitivity studies performed, their operation results in lower margin to acceptance criteria.

For this class of events, it is conservative to delay the reactor trip in order to maximize the mass stored in the RCS at the time of the reactor trip. Hence, pressurizer spray is assumed to operate to avoid an earlier trip on high pressurizer pressure reactor trip setpoint.

The rod control system is assumed to be operating in automatic mode, and thus maintains the average RCS temperature at the initial value even after CMT injection begins. Should the rod control system not be available, the temperature in the RCS would drop following the power decrease dictated by the reactivity effects caused by the CMT boration. An earlier trip on low T_{COLD} would then be actuated with the pressurizer water level well below the high pressurizer water level setpoint.

To summarize, the combination of the above two assumptions result in a longer time to reactor trip and in a larger mass and energy build up in the RCS at the time of the reactor trip and finally in lower margin to pressurizer overfilling. In addition, considering that the loss of offsite power occurs immediately after the trip, no control systems are assumed operable after the trip. Hence, the pressurizer heaters and steam dump are assumed to be not operable.

For the Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory, as reported on page 15.5-7, only pressurizer spray is assumed to operate. The analysis assumptions are taken to maximize the time of reactor trip and to trip the reactor on a "S" signal generated on low T_{COLD}. The S signal also actuates the CMTs that provide an additional mass input to the RCS.

The operation of the spray system prevents the high pressurizer pressure reactor trip setpoint from being reached. If the reactor trip occurs on high pressurizer pressure, following the reactor trip, the CVS would be isolated and CMTs would not be actuated so that the available margin to overfilling would be significantly larger.

If the rod control system is assumed operable together with the spray system, the RCS temperature and pressure would be maintained close to their initial values. Following the CVS injection, the level in the pressurizer would increase and eventually a CVS isolation setpoint on high pressurizer water level would be reached terminating the transient, as described in page 15.5-5.

Therefore, since the operation of the pressurizer spray and automatic rod control result in a more severe transient response, they are assumed to operate in the Chapter 15 accident analysis presented in 15.5.1 and 15.5.2 as described above. They are not "credited" with mitigating the consequences of an accident, and their inclusion in Technical Specification LCO is not appropriate. Note that this approach is also consistent with the AP600 Technical Specifications.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.027

Question:

(Section 16.1, Bases for TS 3.4.12)

A. The Bases Background discussion does not state the size of the Stage 4 automatic depressurization system (ADS) flow path squib and motor operated valves. The staff suggests including this information for completeness.

B. The last sentence of the Bases Background discussion states "The [probabilistic risk assessment] PRA (Ref. 3) shows that adequate core cooling can be provided with the failure of two (or more) flow paths." Is this referring only to the ADS Stage 4 flow paths? What does the "(or more)" mean - that all ADS flow paths can be inoperable without compromising core cooling? Please clarify.

Westinghouse Response:

A. The Bases Background discussions do not include numerical sizes of valves or other components. For example, no sizes are included in the LTOP System (3.4.15), RV Head Vent (3.4.17, Accumulators (3.5.1), CMTs (3.5.2), PRHR HX (3.5.4), and IRWST (3.5.6). No change is proposed.

B. The PRA success criteria for the ADS following a LOCA or a non-LOCA with failure of all other core cooling features is for 3 of 4 ADS stage 4 valves to open. All of the ADS stage 1,2,3 valves can fail to open. This ADS capacity is sufficient to support PXS gravity injection and containment recirculation operation

Design Control Document (DCD) Revision:

Revision to DCD Technical Specifications B 3.4.12.

The number and capacity of the ADS flow paths are selected so that adequate safety injection is provided from the accumulators, IRWST and containment recirculation for the limiting DBA loss of coolant accident (LOCA). For small break LOCAs the limiting single failure is the loss of one fourth stage flow path (Ref. 2). The PRA (Ref. 3) shows that adequate core cooling can be provided with the failure of two (or more) flow paths. The ADS PRA success criteria following a LOCA or non-LOCA with failure of other decay heat removal features is for 3 of 4 ADS stage 4 valves to open. All of the ADS stage 1,2,3 valves can fail to open. This ADS capacity is sufficient to support PXS gravity injection and containment recirculation operation.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

PRA Revision:

None

BASES

BACKGROUND (continued)

stage. The first stage valves can also be modulated to perform a partial RCS depressurization if required. ADS stage 1, 2, 3 valves may be manually operated under controlled conditions for testing purposes.

ADS stages 1, 2 and 3 valves are designed to open relatively slowly, from approximately 25 seconds for the first stage valves, to approximately 70 seconds for the second and third stage valves.

The ADS valves are powered by batteries. In the unlikely event that offsite and onsite AC power is lost for an extended period of time, a timer will actuate ADS within 24 hours of the time at which AC power is lost, before battery power has been degraded to the point where the valves cannot be opened.

The number and capacity of the ADS flow paths are selected so that adequate safety injection is provided from the accumulators, IRWST and containment recirculation for the limiting DBA loss of coolant accident (LOCA). For small break LOCAs the limiting single failure is the loss of one fourth stage flow path (Ref. 2). The PRA (Ref. 3) shows that adequate core cooling can be provided with the failure of two (or more) flow paths. The ADS PRA success criteria following a LOCA or non-LOCA with failure of other decay heat removal features is for 3 of 4 ADS stage 4 valves to open. All of the ADS stage 1, 2, 3 valves can fail to open. This ADS capacity is sufficient to support PXS gravity injection and containment recirculation operation.

APPLICABLE SAFETY ANALYSES

For non-LOCA events, use of the ADS is not required and is not anticipated. For these events, injection of borated water into the core from the CMTs may be required for makeup or boration. However, the amount of water necessary will not reduce the level in the CMTs to the point of ADS actuation.

For events which involve a loss of primary coolant inventory, such as a LOCA, the ADS will be actuated, allowing for injection from the accumulators, the IRWST, and the containment recirculation (Ref. 2).

The ADS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.028

Question:

(Section 16.1, TS 3.4.12 Action A and TS 3.4.13 Action A) Condition A lists three possible conditions. Does each condition get a separate Completion Time? How does this action requirement work? None of the examples in Section 1.3 address this format. Please describe a specific example for each Specification. There may be a clearer way to render these action statement conditions.

Westinghouse Response:

TS 3.4.12 Action A lists three separate Conditions that each have their own 72 hour Completion Time to return the affected flowpath(s) to operable status. TS 3.4.13 Action A lists two separate Conditions that each have their own 72 hour Completion Time to return the affected flowpath(s) to operable status.

The application of the Completion Times as described above for the multiple conditions within a Condition Statement in TS 3.4-12 and TS 3.4-13 (as well as the identified STS TSs above) is consistent with the NRC discussion for the application of Completion Times for this type of Condition Statement as described by the NRC reviewer in Item A of RAI 630.039.

The format of these Actions are consistent with the format of the following STS, Rev. 2 Action statements: TS 3.3.6 Action B; TS 3.4.12 Action G; TS 3.4.16 Action C; TS 3.5.4 Action A.

Since the format is consistent with formats used in the STS, Rev. 2, no change to the AP1000 TS or addition to the examples in Section 1.3 is deemed necessary, in order to prevent deviation from the STS Rev. 2 where there is no technical difference due to the AP1000 plant design.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.029

Question:

(Section 16.1, Bases for TSs 3.4.12, 3.4.13, and 3.4.14) The Bases for each Specification should state which 10 CFR Section 50.36 criterion is satisfied by the ADS system. It also should more clearly state the accidents or transients requiring ADS actuation and the appropriate references.

Westinghouse Response:

The Bases for each Specification will be added; Criterion 3 applies to all three of these Specifications.

The Applicable Safety Analysis section of each of these Specifications will be clarified and appropriate references added. See Below.

Design Control Document (DCD) Revision:

Revision to DCD Technical Specifications Bases, Applicable Safety Analysis for B 3.4.12, B3.4.13, and B 3.4.14.

The ADS satisfies Criterion 3 of the NRC Policy Statement.

Revision to DCD Technical Specifications Bases B 3.4.12, Applicable Safety Analysis.

For non-LOCA events, use of the ADS is not required and is not anticipated. For these events, injection of borated water into the core from the CMTs may be required for makeup or boration. However, the amount of water necessary will not reduce the level in the CMTs to the point of ADS actuation.

For ~~LOCA~~ events which involve a loss of primary coolant inventory, such as a LOCA, ~~it is anticipated that~~ the ADS will be actuated, allowing for injection from the accumulators, the IRWST, and the containment recirculation ~~sump if containment flooding occurs~~ (Ref. 2).

Revision to DCD Technical Specifications Bases B 3.4.13, Applicable Safety Analysis.

For postulated ~~non-LOCA~~ events in MODE 5 with the RCS pressure boundary intact, the primary protection is the Passive Residual Heat Removal Heat Excahanger(PRHR HX).~~.~~ U~~u~~se of the ADS is not required and is not anticipated. For these events, injection of borated water into the core from the core makeup tanks (CMTs) may be required for makeup or boration. However, the amount of water necessary will not reduce the level in the CMTs to the point of ADS actuation.

No LOCAs are postulated during plant operation in MODE 5, however loss of primary coolant through leakage or inadvertent draining may occur. For such ~~postulated~~ shutdown events occurring in MODE 5 it is anticipated that the ADS will be actuated, allowing injection from the in-containment refueling water storage tank (IRWST) and the containment recirculation ~~sump~~ if containment flooding occurs (Ref. 2).

Add Reference 2 to B3.4.13:

REFERENCES
1. AP1000 Probabilistic Risk Assessment, Appendix A.
2. Section 19E.4, "Safety Analyses and Evaluations"

Revision to DCD Technical Specifications Bases B 3.4.14, Applicable Safety Analysis.

When the plant is shutdown with the RCS depressurized, the core makeup tanks (CMTs) are isolated to prevent CMT injection. Since the ADS is actuated by low CMT level, automatic actuation of the ADS is not available. The required ADS stage 1, 2, and 3 vent paths are opened and two ADS stage 4 flow paths are OPERABLE to ensure that in-containment refueling water storage tank (IRWST) injection and containment recirculation can occur, if needed to mitigate events requiring RCS makeup, boration or core cooling (Ref. 1).

The ADS vent path must be maintained until the upper internals are removed, providing an adequate vent path for IRWST injection.

Add Reference 2 to B3.4.14:

REFERENCES ~~None.~~1. Section 19E.4, "Safety Analsyes and Evaluations"

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

PRA Revision:

None

BASES

BACKGROUND (continued)

stage. The first stage valves can also be modulated to perform a partial RCS depressurization if required. ADS stage 1, 2, 3 valves may be manually operated under controlled conditions for testing purposes.

ADS stages 1, 2 and 3 valves are designed to open relatively slowly, from approximately 25 seconds for the first stage valves, to approximately 70 seconds for the second and third stage valves.

The ADS valves are powered by batteries. In the unlikely event that offsite and onsite AC power is lost for an extended period of time, a timer will actuate ADS within 24 hours of the time at which AC power is lost, before battery power has been degraded to the point where the valves cannot be opened.

The number and capacity of the ADS flow paths are selected so that adequate safety injection is provided from the accumulators, IRWST and containment recirculation for the limiting DBA loss of coolant accident (LOCA). For small break LOCAs the limiting single failure is the loss of one fourth stage flow path (Ref. 2). The PRA (Ref. 3) shows that adequate core cooling can be provided with the failure of two (or more) flow paths. The ADS PRA success criteria following a LOCA or non-LOCA with failure of other decay heat removal features is for 3 of 4 ADS stage 4 valves to open. All of the ADS stage 1, 2, 3 valves can fail to open. This ADS capacity is sufficient to support PXS gravity injection and containment recirculation operation.

APPLICABLE SAFETY ANALYSES

For non-LOCA events, use of the ADS is not required and is not anticipated. For these events, injection of borated water into the core from the CMTs may be required for makeup or boration. However, the amount of water necessary will not reduce the level in the CMTs to the point of ADS actuation.

For events which involve a loss of primary coolant inventory, such as a LOCA, the ADS will be actuated, allowing for injection from the accumulators, the IRWST, and the containment recirculation (Ref. 2).

The ADS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The requirement that the 16 ADS valves be OPERABLE ensures that upon actuation, the depressurization of the RCS will proceed smoothly and completely, as assumed in the DBA safety analyses.

(continued)

RAI Number 630.029

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 Automatic Depressurization System (ADS) – Shutdown, RCS Intact

BASES

BACKGROUND	A description of the ADS is provided in the Bases for LCO 3.4.12, "Automatic Depressurization System (ADS) – Operating."
APPLICABLE SAFETY ANALYSES	For postulated events in MODE 5 with the RCS pressure boundary intact, the primary protection is the Passive Residual Heat Removal Heat Exchanger (PRHR HX). Use of the ADS is not required and is not anticipated. For these events, injection of borated water into the core from the core makeup tanks (CMTs) may be required for makeup or boration. However, the amount of water necessary will not reduce the level in the CMTs to the point of ADS actuation. No LOCAs are postulated during plant operation in MODE 5, however loss of primary coolant through leakage or inadvertent draining may occur. For such shutdown events occurring in MODE 5 it is anticipated that the ADS will be actuated, allowing injection from the in-containment refueling water storage tank (IRWST) and the containment recirculation if containment flooding occurs (Ref. 2). The ADS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The requirement that 9 ADS flow paths be OPERABLE assures that upon actuation, the depressurization of the RCS will proceed smoothly and completely, as assumed in the DBA safety analyses. An ADS stage 1, 2, or 3 flow path is considered OPERABLE if both valves in the line are closed and OPERABLE (capable of opening on an actuation signal). In addition, an ADS stage 4 flow path is operable if the motor operated isolation valve is open and the squib valve is closed and OPERABLE (capable of opening on an actuation signal).
APPLICABILITY	In MODE 5 with the reactor coolant pressure boundary (RCPB) intact, 9 flow paths of the ADS must be OPERABLE to mitigate the potential consequences of any event which causes a reduction in the RCS inventory, such as a LOCA.

(continued)

BASES

APPLICABILITY (continued)

The requirements for the ADS in MODES 1 through 4 are specified in LCO 3.4.12, "Automatic Depressurization System (ADS) – Operating;" and in MODE 5 with the RCS pressure boundary open and MODE 6 in LCO 3.4.14, "Automatic Depressurization System (ADS) – Shutdown, RCS Open."

ACTIONS

A.1

If any one, or if two flow paths, consisting of one stage 1 and one stage 2 or 3, are determined to be inoperable, the remaining OPERABLE ADS flow paths are adequate to perform the required safety function. A flow path is inoperable if one or two of the ADS valves in the flow path are determined to be inoperable. A Completion Time of 72 hours is acceptable since the OPERABLE ADS paths can mitigate shutdown events without a single failure.

B.1

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.13 are not met for reasons other than Condition A, the plant must be placed in a MODE in which this LCO does not apply. Action must be initiated, immediately, to place the plant in MODE 5 with the RCS pressure boundary open and ≥ 20% pressurizer level.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

The LCO 3.4.12 Surveillance Requirements (SR 3.4.12.1) are applicable to the ADS valves required to be OPERABLE. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.4.12 for a discussion of each SR.

REFERENCES

1. AP1000 Probabilistic Risk Assessment, Appendix A.

2. Section 19E.4, "Safety Analyses and Evaluations."

RAI Number 630.029

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 Automatic Depressurization System (ADS) – Shutdown, RCS Open

BASES

BACKGROUND

A description of the ADS is provided in the Bases for LCO 3.4.12, "Automatic Depressurization System (ADS) – Operating."

APPLICABLE SAFETY ANALYSES

When the plant is shutdown with the RCS depressurized, the core makeup tanks (CMTs) are isolated to prevent CMT injection. Since the ADS is actuated by low CMT level, automatic actuation of the ADS is not available. The required ADS stage 1, 2, and 3 vent paths are opened and two ADS stage 4 flow paths are OPERABLE to ensure that in-containment refueling water storage tank (IRWST) injection and containment recirculation can occur, if needed to mitigate events requiring RCS makeup, boration or core cooling (Ref. 1).

The ADS vent path must be maintained until the upper internals are removed, providing an adequate vent path for IRWST injection.

The ADS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The requirement that ADS stage 1, 2, and 3 flow paths be open, from the pressurizer through the spargers into the IRWST, and that two ADS stage 4 flow paths be OPERABLE assures that sufficient vent area is available to support IRWST injection.

The Note allows closure of the RCS pressure boundary when the pressurizer level is < 20% to facilitate vacuum refill following mid-loop operations to establish a pressurizer water level ≥ 20%. Prior to closure of the ADS valves, compliance with LCO 3.4.13, ADS – Shutdown, RCS Intact, should be verified.

APPLICABILITY

In MODE 5 with the reactor coolant system pressure boundary (RCPB) open or pressurizer level < 20% and in MODE 6 with the upper internals in place, the stage 1, 2, and 3 ADS vent paths must be open and two ADS stage 4 flow paths be OPERABLE.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Additionally, action to suspend positive reactivity additions is required to ensure that the shutdown margin is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS.

D.1 and D.2

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.14 are not met for reasons other than Conditions A or B while in MODE 6, the plant must be placed in a condition which precludes the need for the ADS vent paths. Action must be initiated, immediately, to remove the upper internals, providing the required vent path. The time to RCS boiling is maximized by increasing RCS inventory and maintaining RCS temperature as low as practical. Additionally, action to suspend positive reactivity additions is required to ensure that the shutdown margin is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Each required ADS flow path is verified to be open by verifying that the stage 1, 2, and 3 valves are in their fully open position every 12 hours, as indicated in the control room. This Surveillance Frequency is acceptable based on administrative controls which preclude repositioning the valves.

SR 3.4.14.2

The LCO 3.4.12 Surveillance Requirements (SR 3.4.12.1 and SR 3.4.12.3) are applicable to the stage 4 ADS valves required to be OPERABLE. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.4.12 for a discussion of each SR.

REFERENCES

1. Section 19E.4, "Safety Analyses and Evaluations."

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.030

Question:

(Section 16.1, TS 1.1 definition of LEAKAGE; and TSs 3.4.16 and 3.4.8; Sections 5.2.5 and 5.2.5.2; and Bases for TS 3.4.11)

A. LCO 3.4.16 and Action A use the term leakage while SR 3.4.16.1 uses the term LEAKAGE. Despite the opening statement of the TS 3.4.16 Bases LCO discussion that RCS pressure isolation valve (PIV) leakage is "identified LEAKAGE," the proposed TS definition of LEAKAGE does not appear to account for RCS PIV leakage (and it is not considered part of "Unidentified LEAKAGE"). Note that DCD Section 5.2.5, second bullet of the last paragraph, states that intersystem leakage is not identified leakage, and refers to TS 3.4.8 Bases. Clarify what is meant by leakage in TS 3.4.16, clearing up inconsistencies among the DCDs, TSs, and Bases. In addition, the Bases for TS 3.4.11 uses LEAKAGE in reference to primary and secondary leak rate; it appears that lower case is appropriate, i.e., leakage.

B. AP1000 SR 3.4.8.1 should be revised to state, "Verify RCS Operational leakage is within limits by performance of RCS water inventory balance." Please revise the DCD accordingly.

Westinghouse Response:

As discussed in the response to RAI 630.001, the AP1000 TSs and Bases have been revised to consistent with Rev. 2 to the NUREG-1431. Therefore, the AP1000 TSs and Bases have been corrected where required so that the use of the words "leakage" and "LEAKAGE" are consistent with their use in STS, Rev. 2.

A. The use of the term "leakage" in the LCO and Action A of TS 3.4.16 is already consistent with the use in the LCO and Action A of STS TS 3.4.14, which was the model for the AP1000 TS 3.4.16.

The word "LEAKAGE" in SR 3.4.16.1 is changed to "leakage" as part of the update to Rev. 2 of the STS, to be consistent with STS SR 3.4.14.1.

The intent of both STS and the AP1000 TSs is that PIV leakage is considered as, and treated as, identified LEAKAGE. Although the definition of LEAKAGE does not explicitly state that "PIV leakage is identified LEAKAGE" in either the STS or the AP1000 TSs, both correctly account for PIV leakage, and the intent to treat PIV leakage as identified LEAKAGE is clearly stated in the Bases for the PIV TS LCO.

The AP1000 LCO 3.4.16 Bases and the STS LCO 3.4.14 Bases both state that "RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS."

PIV leakage meets the criteria for identified leakage in the TS definition, except as stated in the PIV TS LCO Bases, that the leakage is captured to a closed system. Consistent with the TS definition, as identified LEAKAGE, PIV leakage is either captured (in this specific case in a closed system), or if it leaks into the containment atmosphere, comes from a source that is (or would be) specifically located, does not interfere with operation of the leakage detection equipment, and is NOT pressure boundary LEAKAGE.

Therefore, both the AP1000 and STS consistently account for and correctly treat PIV leakage as identified LEAKAGE and no TS or Bases changes are required to address this comment.

The second bullet of the last paragraph of DCD 5.2.5 is revised to delete the word "not," so that this section is consistent with the TS definition for PIV leakage.

The Applicable Safety Analyses section of the Bases for TS 3.4.11 is revised to use lower case "leakage" in reference to primary and secondary leak rate, to be consistent with the use in the corresponding section of the Bases for LCO 3.4.16 in the STS, Rev. 2.

B. SR 3.4.8.1 is updated as requested to be consistent with Rev. 2 of the STS, as part of the response to RAI 630.001.

Design Control Document (DCD) Revision:

DCD TS SR 3.4.16.1

"Verify LEAKAGE of each RCS PIV..." is revised to read "Verify leakage of each RCS PIV..."

See the response to RAI 630.001.

DCD Section 5.2.5, second bullet:

- Leakage into auxiliary systems and secondary systems (intersystem leakage) (This leakage is ~~not~~ considered to be part of the 10 gpm limit identified leakage in the bases of the technical specification 3.4.8. This additional leakage must be considered in the evaluation of the reactor coolant inventory balance.)

See the response to RAI 630.001.

DCD TS 3.4.11, Applicable Safety Analyses Section

"...or indirectly by way of LEAKAGE to the secondary coolant..." is revised to state "...or indirectly by way of leakage to the secondary coolant..."

DCD TS SR 3.4.8.1

"Perform a RCS water inventory balance." was revised to read "Verify RCS Operational leakage is within limits by performance of RCS water inventory balance."

See the response to RAI 630.001.

PRA Revision:

None

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify leakage of each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch valve size up to a maximum of 5 gpm at an RCS pressure ≥ [2215] and ≤ [2255] psig.	24 months

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 RCS Specific Activity

BASES

BACKGROUND

The limits on RCS specific activity ensure that the doses due to postulated accidents are within the doses reported in Chapter 15.

The RCS specific activity LCO limits the allowable concentration of iodines and noble gases in the reactor coolant. The LCO limits are established to be consistent with a fuel defect level of 0.25 percent and to ensure that plant operation remains within the conditions assumed for shielding and Design Basis Accident (DBA) release analyses.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to limit the doses due to postulated accidents to within the values calculated in the radiological consequences analyses (as reported in Chapter 15).

APPLICABLE SAFETY ANALYSES

The LCO limits on the reactor coolant specific activity are a factor in accident analyses that assume a release of primary coolant to the environment either directly as in a Steam Generator Tube Rupture (SGTR) or indirectly by way of leakage to the secondary coolant system and then to the environment (the Steam Line Break).

The events which incorporate the LCO values for primary coolant specific activity in the radiological consequence analysis include the following:

- Steam generator tube rupture (SGTR)
- Steam line break (SLB)
- Locked RCP rotor
- Rod ejection
- Small line break outside containment
- Loss of coolant accident (LOCA) (early stages)

The limiting event for release of primary coolant activity is the SLB. The SLB dose analysis considers the

(continued)

RAI Number 630.030

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	----------NOTE---------- Not required to be performed in MODES 3 and 4 until 12 hours of steady state operation. ---------- Verify RCS Operational leakage is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.8.2	Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.	In accordance with the Steam Generator Tube Surveillance Program

RAI Number 630.030-6

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.031

Question:

(Section 16.1, TS 3.4.8, and Section 5.2.5.3.3)

A. Explain why the primary to secondary LEAKAGE limits in proposed LCO 3.4.8.d and LCO 3.4.8.e have been reduced from the values in the AP600. That is, why were the accident analysis leakage assumptions reduced (see associated Bases discussion),

B. The Frequency Note of SR 3.4.8.1 is redundant and may be deleted,

C. The last sentence of the first paragraph of DCD Section 5.2.5.3.3 contains an extraneous 'and' after the word gaseous.

Westinghouse Response:

A. The secondary LEAKAGE limits have been reduced to 150 gpd/SG in accordance with the guidelines of NEI report NEI-97-06, "Steam Generator Program Guidelines" (Reference 4 of B 3.4.8).

B. SR 3.4.8.1 will be revised as indicated below. NOTE: These changes are also captured in the AP1000 Technical Specification changes to update to STS Revision 2 (see RAI 630.001).

C. The AP1000 DCD will be updated as indicated below.

Design Control Document (DCD) Revision:

From TS 3.4.8, pg. 3.4-13:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	----------NOTE---------- Not required to be performed in MODES 3 and 4 until 12 hours of steady state operation. ----------	~~NOTE Only required to be performed during steady state operation.~~
	~~Perform a~~**Verify RCS operational LEAKAGE is within limits by performance of** RCS water inventory balance.	72 hours
SR 3.4.8.2	Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.	In accordance with the Steam Generator Tube Surveillance Program

From DCD Chapter 5, Section 5.2.5.3.3:

5.2.5.3.3 Containment Atmosphere Radioactivity Monitor

Leakage from the reactor coolant pressure boundary will result in an increase in the radioactivity levels inside containment. The containment atmosphere is continuously monitored for airborne gaseous radioactivity. Air flow through the monitor is provided by the suction created by a vacuum pump. Gaseous ~~and~~ N_{13}/F_{18} concentration monitors indicate radiation concentrations in the containment atmosphere.

PRA Revision:

None

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	----------NOTE---------- Not required to be performed in MODES 3 and 4 until 12 hours of steady state operation. ---------- Verify RCS Operational leakage is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.8.2	Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.	In accordance with the Steam Generator Tube Surveillance Program

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.032

Question:

(Section 16.1, TS 3.4.10)

A. Why does Required Action A.1 not contain a Note that says "LCO 3.0.4 is not applicable."? If it should contain this note, then, consistent with STS 3.4.15, this note should be placed over the Actions table, and removed from the individual Conditions A and B.

B. Why does TS 3.4.10 not adopt STS 3.4.15, Action F, to enter LCO 3.0.3 immediately if all required monitors are inoperable? This action should be included.

C. STS 3.4.15 Required Actions A.1 and B.1.2 to perform RCS water inventory balance (SR 3.4.13.1) contain a note that says "Not required until 12 hours after establishment of steady state operation." If this note applies to the AP1000, it should be adopted in the AP1000 TS 3.4.10 Required Actions A.1 and B.1.2.

Westinghouse Response:

The requested revisions will be included, as shown in the attached mark up.

Design Control Document (DCD) Revision:

Revise Tech. Spec. 3.4.10 as marked in the attachment.

PRA Revision:

None

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 RCS Leakage Detection Instrumentation

LCO 3.4.10 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment sump level channel;

b. One containment atmosphere radioactivity monitor (gaseous N13/F18).

----------NOTE----------
The N13/F18 containment atmosphere radioactivity monitor is only required to be OPERABLE in MODE 1 with RTP > 20%.

APPLICABILITY: MODES 1, 2, 3, and 4.

----------NOTE----------
1. Containment sump level measurements cannot be used for leak detection if leakage is prevented from draining to the sump such as by redirection to the IRWST by the containment shell gutter drains.

2. LCO 3.0.4 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Two containment sump channels inoperable.	A.1 ---------NOTE--------- Not required until 12 hours after establishment of steady state operation. --- Perform SR 3.4.8.1 (RCS inventory balance).	Once per 24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required containment atmosphere radioactivity monitor inoperable.	B.1.1 Analyze grab samples of containment atmosphere.	Once per 24 hours
	OR	
	B.1.2 ---------NOTE--------- Not required until 12 hours after establishment of steady state operation. ----------	
	Perform SR 3.4.8.1.	Once per 24 hours
	AND	
	B.2 Restore containment atmosphere radioactivity monitor to OPERABLE status.	30 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 5.	36 hours
D. All required monitors inoperable.	D.1 Enter LCO 3.0.3.	Immediately

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.033

Question:

(Section 16.1, TS 3.4.11 Bases; TS 3.7.4 Bases; Section 15.6.3.3.1, last paragraph)

The Bases (Applicable Safety Analysis) for STS 3.4.16, "RCS Specific Activity," states that the steam generator tube rupture (SGTR) analysis assumes the secondary coolant DOSE EQUIVALENT I-131 limit of 0.1 µCi/gm (micro-Curies-per-gram). The corresponding discussion in the Bases for the AP1000 TS 3.4.11 does not address assumptions about the secondary coolant activity; however, the last paragraph of DCD 15.6.3.3.1 indicates that SGTR source term assumes 10 percent of the maximum RCS coolant equilibrium activity of iodine and alkali metal activity. Further, the Bases for the AP1000 TS 3.4.11 only mentions the main steam line break, not the SGTR event. Revise the Bases to be more consistent with the STS Bases and the DCD.

Westinghouse Response:

A review of detailed accident analysis results shows that the steam line break (SLB) event is actually the limiting event for release of primary coolant. Therefore, the bases for Technical Specification 3.4.11 will be revised to discuss both SLB and SGTR.

Also, a reference to secondary coolant activities and LCO 3.7.4 will be incorporated.

These revisions are shown in the attached markup.

Design Control Document (DCD) Revision:

Revise TS 3.4.11 as shown in the attachment.

PRA Revision:

None

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 RCS Specific Activity

BASES

BACKGROUND

The limits on RCS specific activity ensure that the doses due to postulated accidents are within the doses reported in Chapter 15.

The RCS specific activity LCO limits the allowable concentration of iodines and noble gases in the reactor coolant. The LCO limits are established to be consistent with a fuel defect level of 0.25 percent and to ensure that plant operation remains within the conditions assumed for shielding and Design Basis Accident (DBA) release analyses.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to limit the doses due to postulated accidents to within the values calculated in the radiological consequences analyses (as reported in Chapter 15).

APPLICABLE SAFETY ANALYSES

The LCO limits on the reactor coolant specific activity are a factor in accident analyses that assume a release of primary coolant to the environment either directly as in a Steam Generator Tube Rupture (SGTR) or indirectly by way of LEAKAGE to the secondary coolant system and then to the environment (the Steam Line Break).

The events which incorporate the LCO values for primary coolant specific activity in the radiological consequence analysis include the following:

Steam generator tube rupture (SGTR)
Steam line break (SLB)
Locked RCP rotor
Rod ejection
Small line break outside containment
Loss of coolant accident (LOCA) (early stages)

The limiting event for release of primary coolant activity is the SLB. The SLB dose analysis considers the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

possibility of a pre-existing iodine spike (in which case the maximum LCO of 60 μCi/gm DOSE EQUIVALENT I-131 is assumed) as well as the more likely initiation of an iodine spike due to the reactor trip and depressurization. In the latter case, the LCO of 1.0 μCi/gm DOSE EQUIVALENT I-131 is assumed at the initiation of the accident, but the primary coolant specific activity is assumed to increase with time due to the elevated iodine appearance rate in the coolant. The reactor coolant noble gas specific activity for both cases is assumed to be the LCO of 280 μCi/gm DOSE EQUIVALENT XE-133. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.4, "Secondary Specific Activity."

The LCO limits ensure that, in either case, the doses reported in Chapter 15 remain bounding.

The RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the specific noble gas activity is limited to 280 μCi/gm DOSE EQUIVALENT XE-133. These limits ensure that the doses resulting from a DBA will be within the values reported in Chapter 15. Secondary coolant activities are addressed by LCO 3.7.4, "Secondary Specific Activity."

The SLB and SGTR accident analyses (Ref. 1 and 2) show that the offsite doses are within acceptance limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SLB or SGTR accident, lead to doses that exceed those reported Chapter 15.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature $\geq$ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity are necessary to contain the potential consequences of a SGTR to within the calculated site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

(continued)

BASES

ACTIONS

C.1 (continued)

RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operation experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

SR 3.4.11.1 requires performing a measure of the noble gas specific activity of the reactor coolant at least once every 7 days. This is a quantitative measure of radionuclides with half lives longer than 15 minutes. This Surveillance provides an indication of any increase in the release of noble gas activity from fuel rods containing cladding defects.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The 7 day Frequency considers the unlikelihood of a significant increase in fuel defect level during the time.

SR 3.4.11.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when increased releases of iodine from the fuel (iodine spiking) is apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level. The Frequency, between 2 and 6 hours after a power change of ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failures; samples at other times would provide inaccurate results.

REFERENCES

1. Section 15.1.5, "Steam System Piping Failure."
2. Section 15.6.3, "Steam Generator Tube Rupture."

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.034

Question:

(Sections 16.1, TS 3.4.17 Bases) The Bases for the reactor vessel head vent system (RVHV) specification does not state which 10 CFR 50.36 criterion applies, and thus, why it is included in the AP1000 TSs. Please revise the Bases to state why this system is in TSs.

Westinghouse Response:

Consistent with the response to AP1000 RAI 630.002, Technical Specification (TS) B 3.4.17 will be revised to specify the Criterion of 10 CFR 50.36 (c) (2) (ii) that is being satisfied. For this RAI, 630.034, Criterion 3 is being satisfied.

Design Control Document (DCD) Revision:

Revise DCD 16.1, TS B 3.4.17 Reactor Vessel Head Vent (RVHV), section entitled, "APPLICABLE SAFETY ANALYSES" as follows:

"For Condition II non-LOCA events, such as inadvertent passive core cooling system operation and chemical and volume control system malfunction, the use of the RVHV may be required to prevent long-term pressurizer overfill (Reference 1).

For LOCA events, the RVHV is not required.

The RVHV satisfies Criterion 3 of 10 CFR 50.36 (c) (2) (ii)."

PRA Revision:

None

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Reactor Vessel Head Vent (RVHV)

BASES

BACKGROUND

The reactor vessel head vent (RVHV) is designed to assure that long-term operation of the Core Makeup Tanks (CMTs) does not result in overfilling of the pressurizer during Condition II Design Basis Accidents (DBAs). The RVHV can be manually actuated by the operators in the main control room to reduce the pressurizer water level during long-term operation of the CMTs.

The RVHV consists of two parallel flow paths each containing two RVHV isolation valves in series. The RVHV valves are connected to the reactor vessel head via a common line. The outlets of the RVHV flow paths combine into one common discharge line which connects to a single ADS discharge header that discharges to spargers located in the incontainment refueling water storage tank (IRWST). The RVHV valves are 1 inch valves with DC solenoid operators.

The RVHV valves are designed to open when actuated by the operator, and to reclose when actuated by the operator from the main control room.

The number and capacity of the RVHV flow paths are selected so that letdown flow from the RCS is sufficient to prevent pressurizer overfill for events where extended operation of the CMTs causes the pressurizer water level to increase. Although realistic evaluations of the Condition II non-LOCA events does not result in pressurizer overfill, conservative analyses of some of these events can result in pressurizer overfill if no operator actions are assumed.

APPLICABLE SAFETY ANALYSES

For Condition II non-LOCA events, such as inadvertent passive core cooling system operation and chemical and volume control system malfunction, the use of the RVHV may be required to prevent long-term pressurizer overfill (Reference 1).

For LOCA events, the RVHV is not required.

The RVHV satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

The requirement that all four RVHV valves be OPERABLE ensures that upon actuation, the RVHV can reduce the pressurizer water level as assumed in the DBA safety analyses.

For the RVHV to be considered OPERABLE, all four valves must be closed and OPERABLE (capable of opening from the main control room).

APPLICABILITY

In MODES 1, 2, 3, and 4 with the RCS not being cooled by the RNS, the RVHV must be OPERABLE to mitigate the potential consequences of any event which causes an increase in the pressurizer water level that could otherwise result in overfilling of the pressurizer.

In MODE 4, with the RCS being cooled by the RNS, and in MODES 5 and 6, operation of the CMTs or CVS will not result in a pressurizer overfill event.

ACTIONS

A.1

If one or two RVHV valves in a single flow path are determined to be inoperable, the flow path is inoperable. The remaining OPERABLE RVHV flow path is adequate to perform the required safety function. A Completion Time of 72 hours is acceptable since the OPERABLE RVHV paths can mitigate DBAs without a single failure.

B.1

If both flow paths are determined to be inoperable, the RVHV is degraded such that the system is not available for some DBA non-LOCA analysis for which may be required. A Completion Time of 6 hours is permitted to restore at least one flow path. This Completion Time is acceptable considering that the realistic analysis of these non-LOCA events do not result in pressurizer overfill.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.17 are not met for reasons other than Conditions A or B, the plant must be brought to MODE 4 where the probability and consequences of an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner, without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

The dedicated component level remote manual valve switches in the main control room shall be used to stroke each RVHV valve to demonstrate OPERABILITY of the controls.

This Surveillance requires verification that each RVHV valve strokes to its fully open position. The Surveillance Frequency for demonstrating valve OPERABILITY references the Inservice Testing Program.

REFERENCES

1. Section 15.5, "Increase in Reactor Coolant System Inventory."

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.035

Question:

(Section 16.1, TS 3.5.1) The Bases Background discussion for the AP1000 TS 3.5.1 states that both accumulators are needed in the event of a large break loss-of-coolant accident (LOCA) caused by a break of a cold leg pipe. Although this is a low probability event, the 8-hour Completion Time for the Condition of one inoperable accumulator does not appear to be justified.

The STS Bases for STS LCO 3.5.1 states that:

"Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 could be violated."

Accordingly, the STS allows just one hour to restore the accumulator to operable status in the event one accumulator is inoperable for reasons other than boron concentration not within limits. In the AP1000 design, the accumulators connect directly to the reactor vessel, which reduces the likelihood that the contents of one accumulator will not flow into the core. A similar arrangement exists in the AP600 design. However, only one accumulator's contents are needed in the AP600. Thus, the 8-hour accumulator restoration Completion Time of the AP600 TS 3.5.1 appears to be justified. It does not appear to be justified for the AP1000, however. Either provide additional justification for the 8-hour Completion Time for the AP1000 design, including additional Bases discussion, or adopt a 1-hour time, consistent with the STS.

Westinghouse Response:

The core melt frequency for large LOCAs in operating plants is typically about 2 E-6 / yr. For AP1000, the large LOCA (pipe breaks) CMF is 4.5E-8/yr, or about 2 orders of magnitude less. This would indicate that there is significantly lower risk from large LOCA's in the AP1000 as compared to operating plants.

Another way of looking at the risk impact of the 8 hour accumulator AOT is contained in Regulatory Guide 1.177 (An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications). This Reg Guide suggests that the risk impact of an AOT change be determined by calculating the Incremental Conditional Core Damage Probability (ICCDP). The ICCDP is calculated by subtracting the baseline CDF from the conditional CDF with the subject equipment out of service and then multiplying by duration of a single AOT under consideration. This Reg Guide states that ICCDP values less than 5.0E-7 are considered to have a small quantitative impact on plant risk. For AP1000, the baseline at-power CDF is 2.41E-7 and the

conditional CDF with one accumulator out of service is 5.20E-6. The following shows the ICCDP for several accumulator AOT's:

	Case 1	Case 2	Case 3	NRC Limit
Accumulator AOT	1 hr	4 hr	8 hr	
ICCDP	5.66E-10	2.26E-9	4.53E-9	5.0E-7

Even at an accumulator AOT of 8 hours, the ICCDP is significantly less than the NRC criteria. This conditional CDF is calculated conservatively by assuming that all large LOCAs would result in core damage if only 1 accumulator is available. In reality, only cold leg large LOCAs would have peak clad temperatures (PCTs) greater than 2200 F. Hot leg large LOCAs would have PCTs well below 2200 F even with one accumulator (see response to RAI 720.010). It is estimated that less than half of the large LOCA initiating event frequency is cold leg LOCAs. In addition, a large cold leg LOCA with a PCT above 2200 F indicates that only a portion of a few rods might be damaged, but a large scale core melt and relocation to the lower reactor vessel head is unlikely.

Another factor in determining this AOT, is what time is sufficient to allow the operators to diagnose the problem and to correct it using proper procedures and checks. It is estimated that an AOT of 8 hours will allow for careful correction of accumulator water levels or pressures and as necessary entry into containment to confirm proper water level / pressure / valve positions. Allowing adequate time for these activities will reduce the chance of mistakes being made during the return of the accumulator to operable status. It will also reduce the possibility of plant shutdowns and the associated risks involved with those operations.

Design Control Document (DCD) Revision:

The following addition is made to the Accumulator TS Bases 3.5.1 to further clarify the justification for the 8 hour AOT

ACTIONS B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 8 hours. With one accumulator inoperable, the remaining accumulator is capable of providing the required safety function, except for one low probability event (large cold leg LOCA) discussed in the background section. The effectiveness of one accumulator is demonstrated in analysis performed to justify PRA success criteria (Ref. 4). The analysis contained in this reference shows that for a range of other events including small LOCAs and large hot leg LOCAs that with one accumulator unavailable the core is adequately cooled. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk (Ref. 6).

The 8 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

The following reference is added to the Accumulator TS Bases 3.5.1 to support the above change:

REFERENCES

1. IEEE Standard 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations."
2. Section 6.3 "Passive Core Cooling System."
3. Section 15.6 "Decrease in Reactor Coolant Inventory."
4. AP1000 PRA Success Criteria.
5. 10 CFR 50.46.
7. Regulatory Guide 1.177, 8/98, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications."

PRA Revision:

None

BASES (continued)

APPLICABILITY

In MODES 1 and 2, and in MODES 3 and 4 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that adequate injection flow from other sources exists to retain peak clad temperatures below the 10 CFR 50.46 limit of 2200°F.

In MODES 3 and 4 with RCS pressure ≤ 1000 psig, and in MODES 5 and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows the RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, action must be taken to restore the parameter.

Deviations in boron concentration are expected to be slight, considering that the pressure and volume are verified once per 24 hours. For one accumulator, boron concentration not within limits will have an insignificant effect on the ability of the accumulators to perform their safety function. Therefore, a Completion Time of 72 hours is considered to be acceptable.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 8 hours. With one accumulator inoperable, the remaining accumulator is capable of providing the required safety function, except for one low probability event (large cold leg LOCA) discussed in the background section. The effectiveness of one accumulator is demonstrated in analysis performed to justify PRA success criteria (Ref. 4). The analysis contained in this reference shows that for a range

(continued)

BASES

ACTIONS

B.1 (continued)

of other events including small LOCAs and large hot leg LOCAs that with one accumulator unavailable the core is adequately cooled. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk (Ref. 7).

The 8 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

C.1 and C.2

If the Required Action and associated Completion Time of Conditions A or B are not met, the plant must be placed in a MODE or condition in which the LCO does not apply. This is done by placing the plant in MODE 3 within 6 hours and with pressurizer pressure to $\leq$ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Each accumulator valve should be verified to be fully open every 12 hours. This verification ensures each accumulator isolation valve is fully open, as indicated in the control room, and timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a partially closed valve could result in not meeting DBA analyses assumptions (Ref. 3). A 12 hour Frequency is considered reasonable in view of the other administrative controls which ensure that a mispositioned isolation valve is unlikely.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.5 (continued)

This SR allows power to be supplied to the motor operated isolation valves when pressurizer pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during unit startup or shutdowns.

Should closure of a valve occur, the safeguard actuation signal provided to the valve would open a closed valve, if required.

SR 3.5.1.6

This SR requires performance of a system performance test of each accumulator to verify flow capabilities. The system performance test demonstrates that the accumulator injection line resistance assumed in accident analyses is maintained. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The System Level Operability Testing Program provides specific test requirements and acceptance criteria.

REFERENCES

1. IEEE Standard 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations."

2. Section 6.3 "Passive Core Cooling System."

3. Section 15.6 "Decrease in Reactor Coolant Inventory."

4. AP1000 PRA Success Criteria.

5. 10 CFR 50.46.

6. NUREG-1366, February 1990.

7. Regulatory Guide 1.177, 8/98, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications."

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.036

Question:

(Section 16.1, TS SR 3.5.1.4) Explain why the AP600 accumulator specified boron concentration range applies to the AP1000, given there is no change in the accumulator specified volume range (SR 3.5.1.3), when there is an increase in reactor pressure vessel (RPV) volume, pressurizer water volume, and steam generator (SG) primary side volume (DCD Sections 5.1 - 5.4) over the AP600 design. Note that the boron concentration remains unchanged for the core makeup tanks (CMTs) while the volume increases by 25 percent (SR 3.5.2.2).

Westinghouse Response:

The AP1000 accumulator boron concentration and water volume did not change from the AP600 values. The T&H analysis performed on the AP1000 as reported in DCD chapter 15 and PRA Appendix A are consistent with the boron concentrations and volumes contained in the AP1000 DCD Technical Specifications. Note that with the lower boron concentrations in the AP1000 RCS, there is a more margin between the accumulator boron concentration and the critical core concentration.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.037

Question:

(Section 16.1, TS 3.5.2 and 3.5.3 Actions)

A. Explain how Action B would apply if Condition B is entered on one CMT on boron concentration and subsequently on the same CMT on temperature, with boron concentration recovery before temperature recovery?

B. Why are there no Conditions addressing Condition A and B together on the same CMT or different CMTs? Also, Condition E should say "One CMT inoperable . . ."

C. The second condition in Action F tries to cover situations not explicitly covered by Actions A through E; however, Section 1.3 allows being in two or more conditions simultaneously as long as the specified conditions together encompass the actual plant condition for the system covered by the LCO. Explain when Action F would apply with multiple other conditions entered simultaneously. Consider revising the Actions to resolve this conflict.

D. Condition A should be revised for clarity as follows: "One CMT inoperable due to one CMT outlet isolation valve inoperable." This also applies to TS 3.5.3 Condition A.

Westinghouse Response:

A. Our intention was to have separate completion times If more than one parameter was out of specification. However, we think that the current wording may not be clear. In order to clarify this situation we will change the wording to what you suggested that we use for the IRWST parameters in RAI 630.039.

B. Conditions A and B occurring on the same CMT are treated separately because their impacts are not additive; condition A affects the reliability of the CMT being actuated whereas condition B affects the margins in the CMT performance once it is actuated. Condition C addresses condition B for both CMTs. If both CMTs become inoperable at the same time for any other reason then condition F part 2 applies.

   Condition E will be changed to say "One CMT inoperable ...".

C. Multiple conditions are considered to be unlikely. However, if they occur within the conditions specified, then the second condition in Action F does not apply, as Section 1.3 allows. The second condition in Action F was included to cover conditions outside of those specified. For example, if one CMT valve was inoperable on both CMTs that would not be covered by Conditions A, B, C, D, and E. In that case the second part of Condition F would apply. No change is proposed.

D, We will change the Condition A wording to say "One CMT inoperable due to ..." in TS 3.5.2. In TS 3.5.3 we will change Condition A to say "Required CMT inoperable due to ...".

Design Control Document (DCD) Revision:

The following changes will be made to TS 3.5.2 Actions.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CMT inoperable due to one CMT outlet isolation valve inoperable.	A.1 Restore outlet isolation valve to OPERABLE status.	72 hours
B. One CMT inoperable due to one or more parameters (water temperature, ~~or~~ boron concentration) not within limits.	B.1 Restore water temperature or boron concentration to within limits.	72 hours
C. Two CMTs inoperable due to water temperature or boron concentration not within limits.	C.1 Restore water temperature or boron concentration to within limits for one CMT.	8 hours
D. One CMT inoperable due to ~~P~~presence of non-condensible gases in one high point vent.	D.1 Vent non-condensible gases.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One CMT inoperable for reasons other than Condition A, B, C, or D.	E.1 Restore CMT to OPERABLE status.	8 hours
F. Required Action or associated Completion Time of Condition A, B, C, D, or E not met. OR LCO not met for reasons other than A, B, C, D, or E.	F.1 Be in MODE 3. AND F.2 Be in MODE 5.	6 hours 36 hours

The following changes will be made to TS 3.5.3 Actions.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required CMT inoperable due to ~~O~~one ~~required~~ outlet isolation valve inoperable.	A.1 Restore required isolation valve to OPERABLE status.	72 hours
B. Required CMT inoperable due to one or more parameters (water temperature, ~~or~~ boron concentration) not within limits.	B.1 Restore water temperature or boron concentration to within limits.	72 hours
C. Required CMT inoperable for reasons other than A or B.	C.1 Restore required CMT to OPERABLE status.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action or associated Completion Time of Conditions A, B, or C not met. OR LCO not met for reasons other than A, B, or C.	D.1 Initiate action to be in MODE 5 with RCS pressure boundary open and ≥ 20% pressurizer level.	Immediately

PRA Revision:

None

RAI Number 630.037

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.2 Core Makeup Tanks (CMTs) – Operating

LCO 3.5.2 Both CMTs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4 with the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CMT inoperable due to one CMT outlet isolation valve inoperable.	A.1 Restore outlet isolation valve to OPERABLE status.	72 hours
B. One CMT inoperable due to one or more parameters (water temperature, boron concentration) not within limits.	B.1 Restore water temperature or boron concentration to within limits.	72 hours
C. Two CMTs inoperable due to water temperature or boron concentration not within limits.	C.1 Restore water temperature or boron concentration to within limits for one CMT.	8 hours
D. One CMT inoperable due to presence of non-condensible gases in one high point vent.	D.1 Vent non-condensible gases.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One CMT inoperable for reasons other than Condition A, B, C, or D.	E.1 Restore CMT to OPERABLE status.	8 hours
F. Required Action and associated Completion Time not met. OR LCO not met for reasons other than A, B, C, D, or E.	F.1 Be in MODE 3. AND F.2 Be in MODE 5.	6 hours 36 hours

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.3 Core Makeup Tanks (CMTs) – Shutdown, RCS Intact

LCO 3.5.3 One CMT shall be OPERABLE.

APPLICABILITY: MODE 4 with the RCS cooling provided by the Normal Residual Heat Removal System (RNS),
MODE 5 with the RCS pressure boundary intact.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required CMT inoperable due to one outlet isolation valve inoperable.	A.1 Restore required isolation valve to OPERABLE status.	72 hours
B. Required CMT inoperable due to one or more parameters (water temperature, boron concentration) not within limits.	B.1 Restore water temperature or boron concentration to within limits.	72 hours
C. Required CMT inoperable for reasons other than A or B.	C.1 Restore required CMT to OPERABLE status.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met. OR LCO not met for reasons other than A, B, or C.	D.1 Initiate action to be in MODE 5 with RCS pressure boundary open and ≥ 20% pressurizer level.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	For the CMT required to be OPERABLE, the SRs of Specification 3.5.2, "Core Makeup Tanks (CMTs) – Operating" are applicable.	In accordance with applicable SRs

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.038

Question:

(Section 16.1, TS 3.5.4 Action B Bases) The Bases should provide more information regarding the importance and function of the IRWST gutter isolation valves. Please revise the DCD accordingly.

Westinghouse Response:

The TS 3.5.4 Bases will be revised to provide discussion of the IRWST gutter isolation valves as shown below.

Design Control Document (DCD) Revision:

Revision to DCD chapter 16.1, TS 3.5.4 Bases Background.

The principle component of the PRHR HX is a 100% capacity heat exchanger mounted in the In-containment Refueling Water Storage Tank (IRWST). The heat exchanger is connected to the Reactor Coolant System (RCS) by a inlet line from one RCS hot leg, and an outlet line to the associated steam generator cold leg channel head. The inlet line to the passive heat exchanger contains a normally open, motor operated isolation valve. The outlet line is isolated by two parallel, normally closed air operated valves, which fail open on loss of air pressure or control signal. There is a vertical collection point at the top of the common inlet piping high point which serves as a gas collector. It is provided with level detectors that indicate when noncondensible gases have collected in this area. There are provisions to manually vent these gases to the IRWST.

In order to preserve the IRWST water for long term PRHR HX operation, a gutter is provided to collect and return water to the IRWST that has condensed on the inside surface of the containment shell. During normal plant operation any water collected by the gutter is directed to the normal containment sump. During PRHR HX operation, redundant series air operated valves are actuated to block the draining of condensate to the normal sump and to force the condensate into the IRWST. These valves fail closed on loss of air pressure or control signal.

The PRHR HX size and heat removal capability is selected to provide adequate core cooling for the limiting non-LOCA heatup Design Basis Accidents (DBAs) (Ref. 2). The Probability Risk Assessment (PRA) (Ref. 3) shows that PRHR HX is not required assuming that passive feed and bleed is available. Passive feed and bleed uses the Automatic Depressurization System (ADS) for bleed and the CMTs/accumulators/IRWST for feed.

PRA Revision:

None

RAI Number 630.038

B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.4 Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating

BASES

BACKGROUND

The normal heat removal mechanism is the steam generators, which are supplied by the startup feedwater system. However, this path utilizes non-safety related components and systems, so its failure must be considered. In the event the steam generators are not available to remove decay heat for any reason, including loss of startup feedwater, the heat removal path is the PRHR HX (Ref. 1).

The principle component of the PRHR HX is a 100% capacity heat exchanger mounted in the In-containment Refueling Water Storage Tank (IRWST). The heat exchanger is connected to the Reactor Coolant System (RCS) by a inlet line from one RCS hot leg, and an outlet line to the associated steam generator cold leg channel head. The inlet line to the passive heat exchanger contains a normally open, motor operated isolation valve. The outlet line is isolated by two parallel, normally closed air operated valves, which fail open on loss of air pressure or control signal. There is a vertical collection point at the top of the common inlet piping high point which serves as a gas collector. It is provided with level detectors that indicate when noncondensible gases have collected in this area. There are provisions to manually vent these gases to the IRWST.

In order to preserve the IRWST water for long term PRHR HX operation, a gutter is provided to collect and return water to the IRWST that has condensed on the inside surface of the containment shell. During normal plant operation any water collected by the gutter is directed to the normal containment sump. During PRHR HX operation, redundant series air operated valves are actuated to block the draining of condensate to the normal sump and to force the condensate into the IRWST. These valves fail closed on loss of air pressure or control signal.

The PRHR HX size and heat removal capability is selected to provide adequate core cooling for the limiting non-LOCA heatup Design Basis Accidents (DBAs) (Ref. 2). The Probability Risk Assessment (PRA) (Ref. 3) shows that PRHR HX is not required assuming that passive feed and bleed is available. Passive feed and bleed uses the Automatic

Depressurization System (ADS) for bleed and the CMTs/accumulators/IRWST for feed.

APPLICABLE SAFETY ANALYSES

In the event of a non-LOCA DBA during normal operation, the PRHR HX is automatically actuated to provide decay heat removal path in the event the normal path through the steam generators is not available (Ref. 2).

The non-LOCA events which establish the PRHR HX parameters are those involving a decrease in heat removal by the

(continued)

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AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.039

Question:

(Section 16.1, TS 3.5.6 Action B and SRs, TSs 3.5.7 and 3.5.8)

A. In Action B, the first condition statement actually contains two separate conditions; hence Condition B is actually three separate Conditions that would each have its own 8-hour Completion Time to return the affected parameter to within limits, and contributing to the return of the IRWST to Operable status. Is this the intent? Suggest all Condition statements like this (where one or more parameters related to component operability are not within limits) be stated similarly to the following:

"IRWST inoperable due to one or more parameters (volume, temperature, and boron concentration) not within limits."

The rules of TS Section 1.3 are clear regarding how to apply this kind of condition statement.

B. If a 3 percent range of values for volume is acceptable, as indicated in the Bases for Required Action B.1, why not specify this range in SR 3.5.6.2? Also, the condition statement "IRWST water volume <100% and >97% of limit" implies that a volume below 100 percent but greater than 97 percent makes the IRWST inoperable. Please clarify.

C. Explain what is meant by the 3 percent given in the second Frequency of SR 3.5.6.3; modify the Frequency statement and the Bases as appropriate to clarify the meaning.

Note, the same or similar three comments apply to TSs 3.5.7 and 3.5.8.

Westinghouse Response:

A. We will revise the condition statement as suggested. Note that we will also revise the Bases to be consistent with the change in the condition statement.

B. The intention was to allow 8 hours to restore the IRWST water volume for small deviations. The deviation is expected to be small because the tank is so large and there are redundant level sensors with alarms set just above the required level. The IRWST water volume limit shown in SR 3.5.6.3 is used in the DCD safety analysis. If the IRWST volume drops slightly below this value, the IRWST is expected to be able to perform its safety functions as discussed in the TS Bases 3.5.6 Actions B.1. This discussion talks about analysis done to support the PRA success criteria determination.

C. The 3% listed in SR 3.5.6.3 is unnecessary and should be deleted. The Bases will also be revised to remove the 3%.

Design Control Document (DCD) Revision:

The following change will be made to TS 3.5.6, 3.5.7 and 3.5.8 Action B.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One motor operated containment recirculation isolation valve not fully open.	A.1 Open motor operated containment recirculation isolation valve.	72 hours
B. IRWST inoperable due to one or more parameters (volume, temperature, and boron concentration) ~~or water temperature~~ not within limits. ~~OR~~ ~~IRWST water volume <100% and >97% of limit.~~	B.1 Restore IRWST to OPERABLE status.	8 hours

The following changes will be made to TS Bases for 3.5.6, 3.5.7 and 3.5.8 Action B.1.

ACTIONS B.1

If the IRWST water volume, boron concentration, or temperature are not within limits, the core cooling capability from injection or PRHR HX heat transfer and the reactivity benefit of injection assumed in safety analyses may not be available. Due to the large volume of the IRWST, online monitoring of volume and temperature, and frequent surveillances, the deviation of these parameters is expected to be minor. ~~The allowable deviation of the water volume is limited to 3%; this limit prevents a significant change in boron concentration and is consistent with~~ The long term cooling analysis performed to justify PRA success criteria

(Ref. 3) assumed multiple failures with as many as 3 CMTs / Accum not injecting. This analysis shows that there is significant margin with respect to the water supplies that support containment recirculation operation. The 8 hour Completion Time is acceptable, considering that the IRWST will be fully capable of performing its assumed safety function in response to DBAs with slight deviations in these parameters.

The following change will be made to TS SR 3.5.6.3 and 3.5.8.3.

SR 3.5.6.3	Verify the IRWST boron concentration is ≥ 2600 ppm and ≤ 2900 ppm.	31 days AND Once within 6 hours after each solution volume increase of 15,000 gal~~., 3.0%~~

The following change will be made to TS Bases for SR 3.5.6.3 and 3.5.8.3 .

Verification every 31 days that the boron concentration of the IRWST is greater than the required limit, ensures that the reactor will remain subcritical following a LOCA. Since the IRWST volume is large and normally stable, the 31 day Frequency is acceptable, considering additional verifications are required within 6 hours after each solution volume increase of 15,000 gal~~., 3%~~. In addition, the relatively frequent surveillance of the IRWST water volume provides assurance that the IRWST boron concentration is not changed.

PRA Revision:

None

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.6 In-containment Refueling Water Storage Tank (IRWST) – Operating

LCO 3.5.6 The IRWST, with two injection flow paths and two containment recirculation flow paths, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One motor operated containment recirculation isolation valve not fully open.	A.1 Open motor operated containment recirculation isolation valve.	72 hours
B. IRWST inoperable due to one or more parameters (volume, temperature, and boron concentration) not within limits.	B.1 Restore IRWST to OPERABLE status.	8 hours
C. One motor operated IRWST isolation valve not fully open.	C.1 Restore motor operated IRWST isolation valve to fully open condition.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met. OR LCO not met for reasons other than A, B, or C.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.6.1	Verify the IRWST water temperature is < 120°F.	24 hours
SR 3.5.6.2	Verify the IRWST borated water volume is > [78,900] cu. ft.	24 hours
SR 3.5.6.3	Verify the IRWST boron concentration is ≥ 2600 ppm and ≤ 2900 ppm.	31 days AND Once within 6 hours after each solution volume increase of 15,000 gal.

(continued)

BASES

ACTIONS (continued)

B.1

If the IRWST water volume, boron concentration, or temperature are not within limits, the core cooling capability from injection or PRHR HX heat transfer and the reactivity benefit of injection assumed in safety analyses may not be available. Due to the large volume of the IRWST, online monitoring of volume and temperature, and frequent surveillances, the deviation of these parameters is expected to be minor. The long term cooling analysis performed to justify PRA success criteria (Ref. 3) assumed multiple failures with as many as 3 CMTs/Accum not injecting. This analysis shows that there is significant margin with respect to the water supplies that support containment recirculation operation. The 8 hour Completion Time is acceptable, considering that the IRWST will be fully capable of performing its assumed safety function in response to DBAs with slight deviations in these parameters.

C.1

If the motor operated IRWST isolation valves are not fully open, injection flow from the IRWST may be less than assumed in the safety analysis. In this situation, the valves must be restored to fully open in 1 hour. This Completion Time is acceptable based on risk considerations.

D.1 and D.2

If the IRWST cannot be returned to OPERABLE status within the associated Completion Times or the LCO is not met for reasons other than Conditions A, B, or C, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.6.1

The IRWST borated water temperature must be verified every 24 hours to ensure that the temperature is within the limit assumed in the accident analysis. This Frequency is sufficient to identify a temperature change that would approach the limit and has been shown to be acceptable through operating experience.

SR 3.5.6.2

Verification every 24 hours that the IRWST borated water volume is above the required minimum level will ensure that a sufficient initial supply is available for safety injection and floodup volume for recirculation and as the heat sink for PRHR. During shutdown with the refueling cavity flooded with water from the IRWST, this Surveillance requires that the combined volume of borated water in the IRWST and refueling cavity meet the specified limit. Since the IRWST volume is normally stable, and is monitored by redundant main control indication and alarm, a 24 hour Frequency is appropriate.

SR 3.5.6.3

Verification every 31 days that the boron concentration of the IRWST is greater than the required limit, ensures that the reactor will remain subcritical following a LOCA. Since the IRWST volume is large and normally stable, the 31 day Frequency is acceptable, considering additional verifications are required within 6 hours after each solution volume increase of 15,000 gal. In addition, the relatively frequent surveillance of the IRWST water volume provides assurance that the IRWST boron concentration is not changed.

SR 3.5.6.4

This surveillance requires verification that each motor operated outlet isolation valve is fully open. This surveillance may be performed with available remote position indication instrumentation. The 12 hour Frequency is acceptable, considering the redundant remote indication and alarms and that power is removed from the valve operator.

(continued)

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.7 In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 5

LCO 3.5.7 The IRWST, with one injection flow path and one containment recirculation flow path, shall be OPERABLE.

APPLICABILITY: MODE 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required motor operated containment recirculation isolation valve not fully open.	A.1 Open required motor operated containment recirculation isolation valve.	72 hours
B. IRWST inoperable due to one or more parameters (volume, temperature, and boron concentration) not within limits.	B.1 Restore IRWST to OPERABLE status.	8 hours
C. Required motor operated IRWST isolation valve not fully open.	C.1 Restore required motor operated IRWST isolation valve to fully open condition.	1 hour

(continued)

BASES (continued)

APPLICABILITY

In MODE 5 with the RCS pressure boundary intact or with the RCS open with pressurizer level ≥ 20%, the IRWST is an RCS injection source of borated water for core cooling and reactivity control. Additionally, in MODE 5 with the RCS pressure boundary intact, the IRWST provides the heat sink for PRHR.

The requirements for the IRWST in MODES 1, 2, 3, and 4 are specified in LCO 3.5.6, In-containment Refueling Water Storage Tank (IRWST) – Operating. The requirements for the IRWST in MODE 6 are specified in LCO 3.5.8, In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 6.

ACTIONS

A.1

If a motor operated containment sump isolation valve in one sump recirculation flow path is not fully open, the valve must be fully opened within 72 hours. The 72 hour Completion Time is consistent with times normally applied to degraded two train ECCS systems which can provide 100% of the required flow without a single failure.

B.1

If the IRWST water volume, boron concentration, or temperature are not within limits, the core cooling capability from injection or PRHR heat transfer and the reactivity benefit of injection assumed in safety analyses may not be available. Due to the large volume of the IRWST, online monitoring of volume and temperature, and frequent surveillances, the deviation of these parameters is expected to be minor. The long term cooling analysis performed to justify PRA success criteria (Ref. 3) assumed multiple failures with as many as 3 CMTs/Accum not injecting. This analysis shows that there is significant margin with respect to the water supplies that support containment recirculation operation. The 8 hour Completion Time is acceptable, considering that the IRWST will be fully capable of performing its assumed safety function in response to shutdown events with slight deviations in these parameters.

(continued)

3.5 PASSIVE CORE COOLING SYSTEMS

3.5.8 In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 6

LCO 3.5.8 The IRWST, with one injection flow path and one containment recirculation flow path, shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required motor operated containment recirculation isolation valve not fully open.	A.1 Open required motor operated containment recirculation isolation valve.	72 hours
B. IRWST and refueling cavity inoperable due to one or more parameters (volume, temperature, and boron concentration) not within limits.	B.1 Restore IRWST to OPERABLE status.	8 hours
C. Required motor operated IRWST isolation valve not fully open.	C.1 Restore required motor operated IRWST isolation valve to fully open condition.	1 hour

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.8.1	Verify the IRWST and refueling cavity water temperature is < 120°F.	24 hours
SR 3.5.8.2	Verify the IRWST and refueling cavity water total borated water volume is > [78,900] cu. ft.	24 hours
SR 3.5.8.3	Verify the IRWST and refueling cavity boron concentration is ≥ 2600 ppm and ≤ 2900 ppm.	31 days AND Once within 6 hours after each solution volume increase of 15,000 gal.
SR 3.5.8.4	For the IRWST and flow paths required to be OPERABLE, the following SRs of Specification 3.5.6, "In-containment Refueling Water Storage Tank (IRWST) – Operating" are applicable: SR 3.5.6.4 SR 3.5.6.6 SR 3.5.6.8 SR 3.5.6.5 SR 3.5.6.7	In accordance with applicable SRs

BASES (continued)

APPLICABILITY

In MODE 6, the IRWST is an RCS injection source of borated water for core cooling and reactivity control.

The requirements for the IRWST in MODES 1, 2, 3, and 4 are specified in LCO 3.5.6, In-containment Refueling Water Storage Tank (IRWST) – Operating. The requirements for the IRWST in MODE 5 are specified in LCO 3.5.7, In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 5.

ACTIONS

A.1

With one motor operated containment sump isolation valve not fully open, the valve must be fully opened within 72 hours. The 72 hour Completion Time is consistent with times normally applied to degraded two train ECCS systems which can provide 100% of the required flow without a single failure.

B.1

If the IRWST and refueling cavity water volume, boron concentration, or temperature are not within limits, the core cooling capability from injection or PRHR HX heat transfer and the reactivity benefit of injection assumed in safety analyses may not be available. Due to the large volume of the IRWST, online monitoring of volume and temperature, and frequent surveillances, the deviation of these parameters is expected to be minor. The long term cooling analysis performed to justify PRA success criteria (Ref. 3) assumed multiple failures with as many as 3 CMTs/Accum not injecting. This analysis shows that there is significant margin with respect to the water supplies that support containment recirculation operation. The 8 hour Completion Time is acceptable considering that the IRWST will be fully capable of performing its assumed safety function in response to shutdown events with slight deviations in these parameters.

C.1

If the motor operated IRWST isolation valves are not fully open, injection flow from the IRWST may be less than assumed in the safety analysis. In this situation, the valves must be restored to fully open in 1 hour. This Completion Time is acceptable based on risk considerations.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.8.2 (continued)

volume of borated water in the IRWST and refueling cavity meet the specified limit. Since the IRWST volume is normally stable, and is monitored by redundant main control indication and alarm, a 24 hour Frequency is appropriate.

SR 3.5.8.3

Verification every 31 days that the boron concentration of the IRWST and refueling cavity is greater than the required limit ensures that the reactor will remain subcritical following shutdown events. Since the IRWST volume is large and normally stable, the 31 day Frequency is acceptable, considering additional verifications are required within 6 hours after each solution volume increase of 15,000 gal.

SR 3.5.8.4

LCO 3.5.6 Surveillance Requirements and Frequencies SR 3.5.6.4 through 3.5.6.8 are applicable to the IRWST and the flow paths required to be OPERABLE. Refer to the corresponding Bases for LCO 3.5.6 for a discussion of each SR.

REFERENCES

None.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.040

Question:

(Section 16.1, SR 3.6.1.1 Note) Regarding the Note in the Frequency column, the Bases do not explain the reason for not allowing the 25 percent Frequency extension of SR 3.0.2. Corresponding STS SR 3.6.1.1 does not contain the note. Revise the Bases to state the reason for the note, if it is even necessary.

Westinghouse Response:

The cited note will be deleted, as shown in the attached mark-up.

Design Control Document (DCD) Revision:

Delete note as shown in attachment.

PRA Revision:

None

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage-rate testing except for containment air-lock testing, in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.041

Question:

(Section 16.1, TS 3.6.3 Actions C and D, and SR 3.6.3.5)

A. Nuclear Energy Institute (NEI)/NRC guidance for STS formatting and style dictate that Condition C not be split between pages as proposed.

B. Delete the unnecessary phrase "in MODES 1, 2, 3, and 4" from Condition D.

C. The surveillance interval for testing automatic CIV actuation on a simulated or actual signal is not necessarily an inservice test. Specify a [18]-month frequency for SR 3.6.3.5 consistent with corresponding STS SR 3.6.3.8. (Choose an interval equal to the cycle duration.)

Westinghouse Response:

A. The Tech Spec will be revised to have the page break before Condition C.

B. "in MODES 1, 2, 3 and 4" will be deleted from Condition D.

C. The surveillance interval will be changed to read "24 months."

Design Control Document (DCD) Revision:

Revised as per attached mark-up.

PRA Revision:

None

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

----------NOTES----------

1. Penetration flow path(s) may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------NOTE-------- Only applicable to penetration flow paths with two containment isolation valves. ---------- One or more penetration flow paths with one containment isolation valve inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND A.2 ---------NOTE---------- 1. Isolation devices in high radiation areas may be verified by use of administrative means.	4 hours

RAI Number 630.041

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
B. --------NOTE-------- Only applicable to penetration flow paths with two containment isolation valves. ---------- One or more penetration flow paths with two containment isolation valves inoperable.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. --------NOTE-------- Only applicable to penetration flow paths with only one containment isolation valve and a closed system. ---------- One or more penetration flow paths with one containment isolation valve inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours
	AND	
	C.2 ---------NOTE---------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. ---------- Verify that the affected penetration flow path is isolated.	Once per 31 days
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each [16 inch] containment purge valve is closed, except when the [16 inch] containment purge valves are open for pressure control, ALARA or air quality considerations for personnel containment entry, or for Surveillances which require the valves to be open.	31 days
SR 3.6.3.2	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. ---------- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	31 days
SR 3.6.3.3	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. ---------- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days

(continued)

RAI Number 630.041

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.5	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	24 months

RAI Number 630.041-6

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.042

Question:

(Section 16.1, TSs 3.6.4, 3.6.5, and 3.6.6 Bases)

In reviewing the AP1000 TS Bases discussions of the Applicable Safety Analyses for TSs 3.6.4, "Containment Pressure," 3.6.5, "Containment Air Temperature," and 3.6.6, "PCS [Passive Containment Cooling System] - Operating," the staff noted that the content of the AP1000 discussion was changed from the content of the corresponding AP600 discussion, which is consistent with STS Revision 2. In Revision 0 to the AP1000 DCD, you removed the safety analysis calculated values but maintained the description and identity of the limiting design-basis accident (DBA), either the LOCA or the steam line break (SLB), as appropriate. In DCD Revision 1, you also removed the description and identity of the limiting event. As a result, the noted TS Bases discussions only indicate that DBAs have been analyzed (in DCD Section 6.2) and that these analyses show the containment design values are not exceeded. Please provide your rationale for these Bases changes or restore the omitted information, consistent with the applicable AP1000 DBA safety analyses.

Westinghouse Response:

The AP1000 TS Bases 3.6.4, 3.6.5, and 3.6.6 have been written so as to include the containment analyses values via reference. This method provides the benefit of TS simplification during plant life as no update to the TS Bases is required solely for the purpose of maintaining the Bases in the event that reanalyses are performed, but remain bounded by the limits identified in the Bases. Thus, the AP1000 DCD 16.1 TS Bases 3.6.4, 3.6.5, and 3.6.6 revision 1 appear somewhat different than their counterparts in AP600, but contain the same information and will reduce future changes to the AP1000 DCD.

The DBA safety analyses results for SLB and LOCA can be found in AP1000 DCD subsection 6.2.1.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.043

Question:

(Section 16.1, TS LCO 3.6.4) The bracket around "≥ -0.2 psig and" should only include "-0.2"; i.e., [-0.2]. Please revise the DCD accordingly.

Westinghouse Response:

The specification will be revised as requested.

Design Control Document (DCD) Revision:

Change Tech Spec 3.6.4 as shown on the attachment.

PRA Revision:

None.

RAI Number 630.043

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ [-0.2] psig and ≤ +1.0 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

Reviewer's Note: The low pressure limit is not needed for plant locations for which the lowest possible ambient temperature is approximately 20°F.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.044

Question:

(Section 16.1, TS 3.6.6 and 3.6.7, LCOs, Actions C and D, and SR 3.6.6.1)

A. Both LCOs 3.6.6 and 3.6.7 should explicitly require three PCS flow paths because the Actions are structured as if there are three operable flow paths. Also, please revise the Bases to make this clear.

B. Condition C should be stated "Water storage tank inoperable due to one or more parameters (volume and temperature) not within limits." This will make the second Condition of Action C less ambiguous. (See AP1000 RAI 630.039.)

C. The note in the Frequency column of SR 3.6.6.1 should be moved to the Surveillance column, and revised to say "Only required to be performed when the ambient temperature is ≤ 32°F or ≥ 100°F."

D. Revise the second paragraph of the Bases for SR 3.6.6.1 to make it clear that the temperature region of 32°F to 100°F is within the **ambient** operating temperature limits of the PCS water storage tank, which ensures the water temperature will remain within the band of 40°F to 120°F, the SR acceptance criteria.

Westinghouse Response:

A. LCO's 3.6.6 and 3.6.7 will be revised as shown in the attached markup. We believe that the current Bases, which were revised as part of DCD Revision 1, are sufficiently clear.

B. The requested revision will be incorporated, as shown in the attached markup.

C. The requested revision will be incorporated, as shown in the attached markup.

D. The paragraph will be revised as shown in the attached markup.

Design Control Document (DCD) Revision:

Revised as attached.

PRA Revision:

None

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Passive Containment Cooling System (PCS) – Operating

BASES

BACKGROUND

The PCS provides containment cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA). The Passive Containment Cooling System is designed to meet the requirements of 10 CFR 50 Appendix A GDC 38 "Containment Heat Removal" and GDC 40 "Testing of Containment Heat Removal Systems" (Ref. 1).

The PCS consists of a 800,000 gal (nominal) cooling water tank, four headered tank discharge lines with flow restricting orifices, and two separate full capacity discharge flow paths to the containment vessel with 3 sets of isolation valves, each capable of meeting the design bases. Algae growth is not expected within the PCCWST; however, to assure water clarity is maintained, a prevailing concentration of hydrogen peroxide is maintained at 50 ppm. The recirculation pumps and heater provide freeze protection for the passive containment cooling water storage tank. However, OPERABILITY of the tank is assured by compliance with the temperature limits specified in SR 3.6.6.1 and not by the recirculation pumps and heater. In addition to the recirculation pumps and heater, the PCS water storage tank temperature can be maintained within limits by the ambient temperature, the large thermal inertia of the tank, or heat from other sources. The PCS valve room temperature must not be below freezing for an extended period to assure the water flow path to the containment shell is available. The isolation valves on each flow path are powered from a separate Division.

Upon actuation of the isolation valves, gravity flow of water from the cooling water tank (contained in the shield building structure above the containment) onto the upper portion of the containment shell reduces the containment pressure and temperature following a DBA. The flow of water to the containment shell surface is initially established to assure that the required short term containment cooling requirements following the postulated worst case LOCA are achieved. As the decay heat from the core becomes less with

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.6 Passive Containment Cooling System (PCS) – Operating

LCO 3.6.6 The passive containment cooling system shall be OPERABLE, with all three water flow paths OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One passive containment cooling water flow path inoperable.	A.1 Restore flow path to OPERABLE status.	7 days
B. Two passive containment cooling water flow paths inoperable.	B.1 Restore flow paths to operable status.	72 hours
C. One or more water storage tank parameters (temperature and volume) not within limits.	C.1 Restore water storage tank to OPERABLE status.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met. OR LCO not met for reasons other than A, B, or C.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	----------NOTE---------- Only required to be performed when the ambient temperature is ≤ 32°F or ≥ 100°F ---------- Verify the water storage tank temperature ≥ 40°F and ≤ 120°F.	24 hours
SR 3.6.6.2	Verify the water storage tank volume ≥ 755,000 gallons.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.3	Verify each passive containment cooling system, power operated, and automatic valve in each flow path that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.6.6.4	Verify each passive containment cooling system automatic valve in each flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.6.6.5	Verify the air flow path from the shield building annulus inlet to the exit is unobstructed and, that all air baffle sections are in place.	24 months
SR 3.6.6.6	Verify passive containment cooling system flow and water coverage performance in accordance with the System Level Operability Testing Program.	At first refueling AND 10 years

3.6 CONTAINMENT SYSTEMS

3.6.7 Passive Containment Cooling System (PCS) – Shutdown

LCO 3.6.7 The passive containment cooling system shall be OPERABLE with all three water flow paths OPERABLE.

APPLICABILITY: MODE 5 with the calculated reactor decay heat > 9.0 MWt,
MODE 6 with the calculated reactor decay heat > 9.0 MWt.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One passive containment cooling water flow path inoperable.	A.1 Restore flow path to OPERABLE status.	7 days
B. Two passive containment cooling water flow paths inoperable.	B.1 Restore flow paths to operable status.	72 hours
C. One or more water storage tank parameters (temperature and volume) not within limits.	C.1 Restore water storage tank to OPERABLE status.	8 hours

(continued)

ACTIONS (continued)

D. Required Action and associated Completion Time not met. OR LCO not met for reasons other than A, B, or C.	D.1.1	If in MODE 5, initiate action to be in MODE 5 with the RCS pressure boundary intact and ≥ 20% pressurizer level.	Immediately
		OR	
	D.1.2	If in MODE 6, initiate action to be in MODE 6 with the water level ≥ 23 feet above the top of the reactor vessel flange.	Immediately
	AND		
	D.2	Suspend positive reactivity additions.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	The SRs of Specification 3.6.6, "Passive Containment Cooling System – Operating" are applicable.	In accordance with applicable SRs

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Passive Containment Cooling System (PCS) – Operating

BASES

BACKGROUND

The PCS provides containment cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA). The Passive Containment Cooling System is designed to meet the requirements of 10 CFR 50 Appendix A GDC 38 "Containment Heat Removal" and GDC 40 "Testing of Containment Heat Removal Systems" (Ref. 1).

The PCS consists of a 800,000 gal (nominal) cooling water tank, four headered tank discharge lines with flow restricting orifices, and two separate full capacity discharge flow paths to the containment vessel with 3 sets of isolation valves, each capable of meeting the design bases. Algae growth is not expected within the PCCWST; however, to assure water clarity is maintained, a prevailing concentration of hydrogen peroxide is maintained at 50 ppm. The recirculation pumps and heater provide freeze protection for the passive containment cooling water storage tank. However, OPERABILITY of the tank is assured by compliance with the temperature limits specified in SR 3.6.6.1 and not by the recirculation pumps and heater. In addition to the recirculation pumps and heater, the PCS water storage tank temperature can be maintained within limits by the ambient temperature, the large thermal inertia of the tank, or heat from other sources. The PCS valve room temperature must not be below freezing for an extended period to assure the water flow path to the containment shell is available. The isolation valves on each flow path are powered from a separate Division.

Upon actuation of the isolation valves, gravity flow of water from the cooling water tank (contained in the shield building structure above the containment) onto the upper portion of the containment shell reduces the containment pressure and temperature following a DBA. The flow of water to the containment shell surface is initially established to assure that the required short term containment cooling requirements following the postulated worst case LOCA are achieved. As the decay heat from the core becomes less with

(continued)

BASES

BACKGROUND (continued)

time, the water flow to the containment shell is reduced in three steps. The change in flow rate is attained without active components in the system and is dependent only on the decreasing water level in the elevated storage tank. In order to ensure the containment surface is adequately and effectively wetted, the water is introduced at the center of the containment dome and flows outward. Weirs are placed on the dome surface to distribute the water and ensure effective wetting of the dome and vertical sides of the containment shell. The monitoring of the containment surface through the Reliability Assurance Program (RAP) and the Inservice Testing Program assures containment surface does not unacceptably degrade containment heat removal performance. During the initial test program, the containment coverage will be measured at the base of the upper annulus in addition to the coverage at the spring line for the full flow case and a lower flow case with PCS recirculation pumps delivering to the containment shell. These benchmark values at the base of the upper annulus will be used to develop acceptance criteria for technical specifications. Contamination can be removed by PCS actuation and by using coating vendor cleaning procedures.

The path for the natural circulation of air is from the air intakes in the shield building, down the outside of the baffle, up along the containment shell to the top, center exit in the shield building and is always open. The drains in the upper annulus region must be clear to prevent water from blocking the air flow path. Heat is removed from within the containment utilizing the steel containment shell as the heat transfer surface combining conductive heat transfer to the water film, convective heat transfer from the water film to the air, radiative heat transfer from the film to the air baffle, and mass transfer (evaporation) of the water film into the air. As the air heats up and water evaporates into the air, it becomes less dense than the cooler air in the air inlet annulus. This differential causes an increase in the natural circulation of the air upward along the containment surface, with heated air/water vapor exiting the top/center of the shield building. Additional system design details are provided in reference 3.

The PCS is actuated either automatically, by a containment High-2 pressure signal, or manually. Automatic actuation opens the cooling water tank discharge valves, allowing gravity flow of the cooling water onto the containment shell. The manual containment cooling actuation consists of

(continued)

BASES

BACKGROUND (continued)

four momentary controls, if two associated controls are operated simultaneously actuation will occur in all divisions. The discharge continues for at least three days.

The PCS is designed to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a DBA.

The PCS is an ESF system and is designed to ensure that the heat removal capability required during the post accident period can be attained.

APPLICABLE SAFETY ANALYSES

The Passive Containment Cooling System limits the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF system, assuming the loss of one Class 1E Engineered Safety Features Actuation Cabinet (ESFAC) Division, which is the worst case single active failure and results in one PCS flow path being inoperable.

The analyses and evaluations assume a unit specific power level of 3400 MWt, one passive containment cooling train operating, and initial (pre-accident) containment conditions of 120°F and 1.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Passive Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

Containment cooling system performance for post accident conditions is given in Reference 3. The result of the analysis is that each train can provide 100% of the required peak cooling capacity during the post accident condition.

The modeled Passive Containment Cooling System actuation response time from the containment analysis is based upon a response time associated with exceeding the containment High-2 pressure setpoint to opening of isolation valves.

The Passive Containment Cooling System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, one passive containment cooling water flow path is required to maintain the containment peak pressure and temperature below the design limits (Ref. 3). To ensure that this requirement is met, two passive containment cooling water flow paths are provided. Therefore, in the event of an accident, at least one flow path operates, assuming the worst case single active failure occurs. A third PCS flow path is provided for protection against multiple failure scenarios modeled in the PRA. To ensure that these requirements are met, three PCS water flow paths must be OPERABLE.

The PCS includes a cooling water tank, valves, piping, instruments and controls to ensure an OPERABLE flow path capable of delivering water from the cooling water tank upon an actuation signal. An OPERABLE flow path consists of a normally closed valve capable of automatically opening in series with a normally open valve. For the two flow paths containing air-operated valves, it is preferred because of PRA insights that these valves be normally closed.

The PCS cooling water storage tank ensures that an adequate supply of water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA). To be considered OPERABLE, the PCS cooling water storage tank must meet the water volume and temperature limits

(continued)

BASES

LCO (continued)

established in the SRs. To be considered OPERABLE, the air flow path from the shield building annulus inlet to the exit must be unobstructed, with unobstructed upper annulus safety-related drains providing a path for containment cooling water runoff to preclude blockage of the air flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the PCS.

During shutdown the PCS may be required to remove heat from containment. The requirements in MODES 5 and 6 are specified in LCO 3.6.7, Passive Containment Cooling System (PCS) – Shutdown.

ACTIONS

A.1

With one passive containment cooling water flow path inoperable, the affected flow path must be restored within 7 days. In this degraded condition, the remaining flow paths are capable of providing greater than 100% of the heat removal needs after an accident, even considering the worst single failure. The 7 day Completion Time was chosen in light of the remaining heat removal capability and the low probability of a DBA occurring during this period.

B.1

With two passive containment cooling water flow paths inoperable, at least one affected flow path must be restored to OPERABLE status within 72 hours. In this degraded condition, the remaining flow path is capable of providing greater than 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen in light of the remaining heat removal capability and the low probability of DBA occurring during this period.

C.1

If the cooling water tank is inoperable, it must be restored to OPERABLE status within 8 hours. The tank may be declared

(continued)

BASES

ACTIONS

C.1 (continued)

inoperable due to low water level or temperature out of limits. The 8 hour Completion Time is reasonable based on the remaining heat removal capability of the system and the availability of cooling water from alternate sources.

D.1 and D.2

If any of the Required Actions and associated Completion Times for Condition A or B are not met, or if the LCO is not met for reasons other than Condition A or B, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

This surveillance requires verification that the cooling water temperature is within the limits assumed in the accident analyses. The 24 hour Frequency is adequate to identify a temperature change that would approach the temperature limit and has been shown to be acceptable in similar applications.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are between 32°F and 100°F, which ensures that the PCS tank water temperature remains within the limits specified in SR 3.6.6.1.

SR 3.6.6.2

Verification that the cooling water volume is above the required minimum ensures that a sufficient supply is available for containment cooling. Since the cooling water

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.2 (continued)

volume is normally stable and low level is indicated by a main control room alarm, a 7 day Frequency is appropriate and has been shown to be acceptable in similar applications.

SR 3.6.6.3

Verifying the correct alignment of power operated, and automatic valves, excluding check valves, in the Passive Containment Cooling System provides assurance that the proper flow paths exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct positions prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through control room instrumentation or a system walkdown, that valves capable of potentially being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single flow path. This Frequency has been shown to be acceptable through operating experience.

SR 3.6.6.4

This SR requires verification that each automatic isolation valve actuates to its correct position upon receipt of an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirmed by operating experience) of the equipment. Operating experience has shown that these components usually pass the Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.5

Periodic inspections of the PCS air flow path from the shield building annulus inlet to the exit ensure that it is unobstructed, the baffle plates are properly installed, and the upper annulus safety-related drains are unobstructed. Although there are no anticipated mechanisms which would cause air flow path or annulus drain obstruction and the effect of a missing air baffle section is small, it is considered prudent to verify this capability every 24 months. Additionally, the 24 month Frequency is based on the desire to perform this Surveillance under conditions that apply during a plant outage, on the need to have access to the locations, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation in similar situations.

SR 3.6.6.6

This SR requires performance of a Passive Containment Cooling System test to verify system flow and water coverage capabilities. The system performance test demonstrates that the containment cooling capability assumed in accident analyses is maintained by verifying the flow rates via each standpipe and measurement of containment wetting coverage. The System Level Operability Testing Program provides specific test requirements and acceptance criteria. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The first refueling and 10 year Frequency is based on the ability of the more frequent surveillances to verify the OPERABILITY of the active components and features which could degrade with time.

REFERENCES

1. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

2. 10 CFR 50, Appendix K, "ECCS Evaluation Models."

3. Chapter 6.2, "Containment Systems."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Passive Containment Cooling System (PCS) – Shutdown

BASES

BACKGROUND	A description of the PCS is provided in the Bases for LCO 3.6.6, "Passive Containment Cooling System – Operating."
APPLICABLE SAFETY ANALYSES	The PCS limits the temperature and pressure that could be experienced during shutdown following a loss of decay heat removal. For shutdown events, the Reactor Coolant System (RCS) sensible and decay heat removal requirements are reduced as compared to heat removal requirements for MODE 1, 2, 3, or 4 events. Therefore, the shutdown containment heat removal requirements are bounded by analyses of MODES 1, 2, 3, and 4 events. A discussion of MODES 1, 2, 3, and 4 DBAs is provided in the Bases for LCO 3.6.6, "Passive Containment Cooling System (PCS) – Operating." The PCS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	For postulated shutdown events, one passive containment cooling water flow path is required to provide the required containment heat removal capability (Ref. 1). To ensure that this requirement is met, two passive containment cooling water flow paths are provided. Therefore, in the event of an accident, at least one flow path operates, assuming the worst case single active failure occurs. A third PCS flow path is provided for protection against multiple failure scenarios modeled in the PRA. To ensure that these requirements are met, three PCS water flow paths must be OPERABLE. The PCS includes a cooling water tank, valves, piping, instruments and controls to ensure an OPERABLE flow path capable of delivering water from the cooling water tank upon an actuation signal. The PCS cooling water storage tank ensures that an adequate supply of water is available to cool and depressurize the containment in the event of a loss of decay heat removal. To be considered OPERABLE, the PCS cooling water storage

(continued)

BASES

LCO (continued)

tank must meet the water volume and temperature limits established in the SRs. To be considered OPERABLE, the air flow path from the shield building annulus inlet to the exit must be unobstructed, with unobstructed upper annulus safety-related drains providing a path for containment cooling water runoff to preclude blockage of the air flow path.

APPLICABILITY

OPERABILITY of the PCS is required in either MODE 5 or 6 with the calculated reactor decay heat greater than 9 MWt for heat removal in the event of a loss of nonsafety decay heat removal capabilities.

With the decay heat less than 9 MWt, the decay heat can be easily removed from containment with air cooling alone. Confirmation of decay heat levels may be determined consistent with the assumptions and analysis basis of ANS 1979 plus 2 sigma or via an energy balance of the reactor coolant system.

The PCS requirements in MODES 1, 2, 3, and 4 are specified in LCO 3.6.6, Passive Containment Cooling System (PCS) – Operating.

ACTIONS

A.1

With one passive containment cooling water flow path inoperable, the affected flow path must be restored within 7 days. In this degraded condition, the remaining flow paths are capable of providing greater than 100% of the heat removal needs after an accident, even considering the worst single failure. The 7 day Completion Time was chosen in light of the remaining heat removal capability and the low probability of a DBA occurring during this period.

B.1

With two passive containment cooling water flow paths inoperable, at least one affected flow path must be restored to OPERABLE status within 72 hours. In this degraded condition, the remaining flow path is capable of providing greater than 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen in light of the remaining heat removal capability and the low probability of an event occurring during this period.

(continued)

BASES

ACTIONS (continued)

C.1

If the cooling water tank is inoperable, it must be restored to OPERABLE status within 8 hours. The tank may be declared inoperable due to low water volume or temperature out of limits. The 8 hour Completion Time is reasonable based on the remaining heat removal capability of the system and the availability of cooling water from alternate sources.

D.1.1, D.1.2, and D.2

Action must be initiated if any of the Required Actions and associated Completion Times for Condition A or B are not met, or if the LCO is not met for reasons other than Condition A or B. If in MODE 5 with the RCS pressure boundary open and/or pressurizer level < 20%, action must be initiated, immediately, to increase the RCS level to a pressurizer level ≥ 20% and to close the RCS so that the PRHR HX operation is available. If in MODE 6, action must be initiated, immediately, to increase the refueling cavity water level ≥ 23 feet above the top of the reactor vessel flange. In both cases, the time to RCS boiling is maximized by maximizing the RCS inventory and maintaining RCS temperature as low as practical. Additionally, action to suspend positive reactivity additions is required to ensure that the shutdown margin is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS.

These Actions place the plant in a condition which maximize the time to actuation of the Passive Containment Cooling System, thus providing time for repairs or application of alternative cooling capabilities.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

The LCO 3.6.6 Surveillance Requirements (SR 3.6.6.1 through 3.6.6.6) are applicable. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.6.6 for a discussion of each SR.

REFERENCES

1. Section 6.2, "Containment Systems."

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.045

Question:

(Section 16.1, TS 3.6.8 Bases) The Bases for TS 3.6.8 reference the PRA descriptions in DCD Chapter 19 and the AP1000 PRA, but do not explicitly state the specific bounding postulated shutdown event(s). The Background discussion, fifth paragraph, indicates that events leading to loss of RNS cooling is the main concern. To minimize loss of cooling water inventory for such events, containment penetrations allowed by the LCO to be open should be closed before steaming occurs, within a time frame consistent with Figure B 3.6.8-1. However, for clarity and completeness, please: (a) Revise the Bases to explicitly state what the postulated events are, (b) clarify what is meant by "such events" (according to the last sentence of the fifth paragraph, the hot leg level instrument function to automatically isolate letdown on hot leg low level precludes loss of RNS cooling for "such events "), and (c) state which of the Criteria of 10 CFR 50.36 apply to this LCO.

Westinghouse Response:

The fifth paragraph of the basis for Tech Spec 3.6.8 will be revised as shown in the attached mark-up. This addresses points (a) and (b) above.

10 CFR 50.36(c)(2)(ii) applies to this LCO. This is now shown in the "Applicable Safety Analyses" section of the attached markup. This addresses point (c) above. It is noted that this change is also addressed in the response to AP1000 RAI 630.002.

Design Control Document (DCD) Revision:

As shown in the attached markup.

PRA Revision:

None

Insert 1

As presented in Chapter 54 of Reference 2, the most risk significant events during shutdown are events that lead to a loss of RNS cooling. Of these, the limiting events that lead to steaming to containment are the loss of shutdown cooling events, specifically:

- Loss of decay heat removal during drained conditions due to a failure of component cooling water or service water system;
- Loss-of-offsite power during drained conditions; and
- Loss of decay heat removal during drained conditions due to failure of the normal residual heat removal system.

These events are further discussed in section 19.59.5 of Reference 1. Time to steaming is dependent on the postulated RCS configuration (intact versus open), and is based on the response of the plant considering features such as the operation of the 4th stage ADS valves if necessary, status of the upper internals, status of refueling cavity, etc. Conservative assumptions regarding these features are made in the determination of the minimum time to steaming.

The risk of overdraining the RCS has been significantly reduced in the AP1000 due to the automatic protection features associated with the hot leg level instruments which isolate letdown on low hot leg water level. Since overdraining the RCS is no longer a significant contributor to core damage, as shown in Table 54-4 of Reference 2.

RAI Number 630.045

BASES

BACKGROUND (continued)

steaming may be different than that provided in Figure B 3.6.8-1. In determining the minimum time to steaming, conservative assumptions regarding core decay heat, RCS configuration, and initial RCS inventory are used to minimize the calculated time to steaming.

As presented in Tables 54-1 and 54-4 of Reference 2, the most risk significant events during shutdown are events that lead to a loss of RNS cooling. Of these, the limiting events that lead to steaming to containment are the loss of shutdown cooling events, specifically:

- Loss of decay heat removal during drained conditions due to a failure of component cooling water or service water system;
- Loss-of-offsite power during drained conditions; and
- Loss of decay heat removal during drained conditions due to failure of the normal residual heat removal system.

These events are further discussed in Section 19.59.5 of Reference 1. Time to steaming is dependent on the postulated RCS configuration (intact versus open), and is based on the response of the plant considering features such as the operation of the 4th stage ADS valves if necessary, status of the upper internals, status of refueling cavity, etc. Conservative assumptions regarding these features are made in the determination of the minimum time to steaming.

The risk of overdraining the RCS has been significantly reduced in the AP1000 due to the automatic protection features associated with the hot leg level instruments which isolate letdown on low hot leg water level. Since a loss of RNS cooling does not occur for this event it is no longer a significant contributor to core damage, as shown in Table 54-4 of Reference 2.

The assumptions used in determining the required closure time for the various containment openings should be conservative, and should be consistent with the plant operating procedures, staffing levels, and status of the containment openings. The evaluation should consider the ability to close the containment for the limiting loss of shutdown cooling event, and considering the possibility of a station blackout.

RAI Number 630.045-3

The containment equipment hatches, which are part of the containment pressure boundary, provide a means for moving large equipment and components into and out of containment. If closed, the equipment hatch must be held in place by at least [four] bolts. Good engineering practice dictates that bolts required by this LCO be approximately equally spaced. Alternatively, if open, each equipment hatch can be installed using a dedicated set of hardware, tools and equipment. A self-contained power source is provided to drive each hoist while lowering the hatch into position. Large equipment and components may be moved through the hatches as long as they can be removed and the hatch closed prior to steaming into the containment.

(continued)

RAI Number 630.045

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AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.046

Question:

(Section 16.1, omitted TS 3.6.10) Please explain why TSs for the hydrogen recombiners are not included.

Westinghouse Response:

Westinghouse prepared the AP1000 DCD in anticipation of the revised 10 CFR 50.44. The proposed 10 CFR 50.44 that has been published in the Federal Register (FR Volume 67, No. 149, pg. 50374) for comments permits the systems, components and instrumentation that perform the post-accident hydrogen monitoring functions to be classified as non-safety related. In addition, the hydrogen control systems whose function is based on controlling hydrogen buildup for design basis accidents may be eliminated from the plant design. Only those control systems required to protect the containment from severe accidents must be retained.

As a result, Westinghouse has eliminated TS 3.6.10 for the hydrogen recombiners. This is consistent with the staff recommendations associated with the revised 10 CFR 50.44.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.047

Question:

(Section 16.1, TS Section 3.7) Acceptance of TS Section 3.7 requires resolution of RAIs 410.008, 410.009, and 410.013. Please address these issues and revise the DCD/TSs accordingly.

NOTE: AP1000 RAIs 410.008, 410.009, and 410.013 were issued on September 30, 2002 (ADAMS Accession No. ML022680338).

Westinghouse Response:

There are no Technical Specification changes as a result of reviewing RAIs 410.008, 410.009 and 410.013.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.048

Question:

(Section 16.1, TS 3.7.5) Please change the Applicability of TS 3.7.5 to be consistent with corresponding STS 3.7.15.

Westinghouse Response:

Technical Specifications 3.7.5 and 3.7.15 address spent fuel pool level in the AP1000 and the Standard Technical Specifications, respectively. The applicability for each specification is given below.

> AP1000 LCO 3.7.5 – Applicability: At all times.
>
> Standard Technical Specifications LCO 3.7.15 – Applicability: During movement of irradiated fuel assemblies in the fuel storage pool.

In the AP1000, the required minimum water level in the spent fuel pool provides the large capacity safety-related heat sink in the event the normal spent fuel pool cooling system is inoperable. Therefore, in the AP1000, the minimum required water level should be maintained at all times rather than only when irradiated fuel assemblies are being moved in the pool.

Therefore, the applicability for the AP1000 spent fuel pool water level specification (LCO 3.7.5) remains as currently shown.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.049

Question:

(Section 16.1, TS 3.7.6) It is recommended that you adopt TSTF 287, Revision 5 for TS 3.7.6, consistent with STS TS 3.7.10. Although the AP600 has a 24-hour allowed outage time (AOT) (Action C) for an inoperable control room boundary, the handling of this provision, including Notes and Bases discussions related to intentionally making the boundary inoperable, was incomplete, and should be improved in the AP1000 TSs (consistent with TSTF 287 and Revision 2 to the STS).

Westinghouse Response:

In accordance with RAI 630.001, TS 3.7.6 and its Bases are being updated, as shown in the attachments, to be consistent with STS, Rev. 2.

Design Control Document (DCD) Revision:

See attachments.

PRA Revision:

None

RAI Number 630.049

3.7 PLANT SYSTEMS

3.7.6 Main Control Room Habitability System (VES)

LCO 3.7.6 The Main Control Room (MCR) Habitability System shall be OPERABLE.

----------NOTE----------
The MCR boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

----------NOTE----------
LCO 3.0.8 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One VES valve or damper inoperable.	A.1 Restore VES valve or damper to OPERABLE status.	7 days
B. MCR air temperature not within limit.	B.1 Restore MCR air temperature to within limit.	24 hours
C. Loss of integrity of MCR pressure boundary.	C.1 Restore MCR pressure boundary to OPERABLE status.	24 hours
D. Required Action and associated Completion Time of Conditions A, B, or C not met in MODE 1, 2, 3, or 4.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

E. Required Action and associated Completion Time of Conditions A, B, or C not met during movement of irradiated fuel.	E.1 Suspend movement of irradiated fuel assemblies.	Immediately

(continued)

RAI Number 630.049

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RAI Number 630.049-4

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. VES inoperable in MODE 1, 2, 3, or 4.	F.1 Be in MODE 3. AND F.2 Be in MODE 4. AND F.3 Restore VES to OPERABLE status.	6 hours 12 hours 36 hours
G. VES inoperable during movement of irradiated fuel.	G.1 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify Main Control Room air temperature is ≤ 75°F.	24 hours
SR 3.7.6.2	Verify that the compressed air storage tanks are pressurized to [≥ 3400 psig].	24 hours
SR 3.7.6.3	Verify that each VES air delivery isolation valve is OPERABLE.	In accordance with the Inservice Testing Program
SR 3.7.6.4	Verify that each VES air header manual isolation valve is in an open position.	31 days
SR 3.7.6.5	Verify that the air quality of the air storage tanks meets the requirements of Appendix C, Table C-1 of ASHRAE Standard 62.	92 days
SR 3.7.6.6	Verify that all VBS Main Control Room isolation valves are OPERABLE and will close upon receipt of an actual or simulated actuation signal.	24 months
SR 3.7.6.7	Verify that each VES pressure relief isolation valve within the MCR pressure boundary is OPERABLE.	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.6.8	Verify that each VES pressure relief damper is OPERABLE.	24 months
SR 3.7.6.9	Verify that the self contained pressure regulating valve in each VES air delivery flow path is OPERABLE.	In accordance with the Inservice Testing Program
SR 3.7.6.10	Verify that one VES air delivery flow path maintains a 1/8 inch water gauge positive pressure in the MCR envelope relative to the adjacent areas at the required air addition flow rate of 65 ± 5 scfm using the safety related compressed air emergency air storage tanks.	24 months

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Control Room Emergency Habitability System

BASES

BACKGROUND

The Main Control Room Habitability System (VES) provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. The system is designed to operate following a Design Basis Accident (DBA) which requires protection from the release of radioactivity. In these events, the Nuclear Island Non-Radioactive Ventilation System (VBS) would continue to function if AC power is available. If AC power is lost or a High-2 main control room (MCR) radiation signal is received, the VES is actuated. The major functions of the VES are: 1) to provide forced ventilation to deliver an adequate supply of breathable air for the MCR occupants; 2) to provide forced ventilation to maintain the MCR at a 1/8 inch water gauge positive pressure with respect to the surrounding areas; and 3) to limit the temperature increase of the MCR equipment and facilities that must remain functional during an accident, via the heat absorption of passive heat sinks.

The VES consists of compressed air storage tanks, two air delivery flow paths, associated valves, piping, and instrumentation. The tanks contain enough breathable air to supply the required air flow to the MCR for at least 72 hours. The VES system is designed to maintain CO_2 concentration less than 0.5% for up to 11 MCR occupants.

Sufficient thermal mass exists in the surrounding concrete structure (including walls, ceiling and floors) to absorb the heat generated inside the MCR, which is initially at or below 75°F. Heat sources inside the MCR include operator workstations, emergency lighting and occupants. Sufficient insulation is provided surrounding the MCR pressure boundary to preserve the minimum required thermal capacity of the heat sink. The insulation also limits the heat gain from the adjoining areas following the loss of VBS cooling.

In the unlikely event that power to the VBS is unavailable for more than 72 hours, MCR envelope habitability is maintained by operating one of the two MCR ancillary fans to supply outside air to the MCR envelope.

(continued)

RAI Number 630.049

BASES

BACKGROUND (continued)

The compressed air storage tanks are initially pressurized to 3400 psig. During operation of the VES, a self contained pressure regulating valve maintains a constant downstream pressure regardless of the upstream pressure. An orifice downstream of the regulating valve is used to control the air flow rate into the MCR. The MCR is maintained at a 1/8 inch water gauge positive pressure to minimize the infiltration of airborne contaminants from the surrounding areas.

APPLICABLE SAFETY ANALYSES

The compressed air storage tanks are sized such that the set of tanks has a combined capacity that provides at least 72 hours of VES operation.

Operation of the VES is automatically initiated by either of two safety related signals: 1) undervoltage to Class 1E battery charger, or 2) high-2 particulate or iodine radioactivity.

In the event of a loss of all AC power, the VES functions to provide ventilation, pressurization, and cooling of the MCR pressure boundary.

In the event of a high level of gaseous radioactivity outside of the MCR, the VBS continues to operate to provide pressurization and filtration functions. The MCR air supply downstream of the filtration units is monitored by a safety related radiation detector. Upon an undervoltage to Class 1E battery charger or high-2 particulate or iodine radioactivity setpoint, a safety related signal is generated to isolate the MCR from the VBS and to initiate air flow from the VES storage tanks. Isolation of the VBS consists of closing safety related valves in the supply and exhaust ducts that penetrate the MCR pressure boundary. VES air flow is initiated by a safety related signal which opens the isolation valves in the VES supply lines.

The VES functions to mitigate a DBA or transient that either assumes the failure of or challenges the integrity of the fission product barrier.

The VES satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

RAI Number 630.049

BASES (continued)

LCO

The VES limits the MCR temperature rise and maintains the MCR at a positive pressure relative to the surrounding environment.

Two air delivery flow paths are required to be OPERABLE to ensure that at least one is available, assuming a single failure.

The VES is considered OPERABLE when the individual components necessary to deliver a supply of breathable air to the MCR are OPERABLE. This includes components listed in SR 3.7.6.2 through 3.7.6.8. In addition, the MCR pressure boundary must be maintained, including the integrity of the walls, floors, ceilings, electrical and mechanical penetrations, and access doors.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

The LCO is modified by a Note allowing the control room boundary to be opened intermittently under administrative controls. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

APPLICABILITY

The VES is required to be OPERABLE in MODES 1, 2, 3, and 4 and during movement of irradiated fuel because of the potential for a fission product release following a DBA.

The VES is not required to be OPERABLE in MODES 5 and 6 when irradiated fuel is not being moved because accidents resulting in fission product release are not postulated.

ACTIONS

LCO 3.0.8 is applicable while in MODE 5 or 6. Since irradiated fuel assembly movement can occur in MODE 5 or 6, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, the fuel movement is independent of shutdown reactor operations. Entering

LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1

When a VES valve or damper is inoperable, action is required to restore the component to OPERABLE status. A Completion Time of 7 days is permitted to restore the valve or damper to OPERABLE status before action must be taken to reduce power. The Completion Time of 7 days is based on engineering judgment, considering the low probability of an accident that would result in a significant radiation release from the fuel, the low probability of not containing the radiation, and that the remaining components can provide the required capability.

(continued)

RAI Number 630.049

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BASES

ACTIONS (continued)

B.1

When the main control room air temperature is outside the acceptable range during VBS operation, action is required to restore it to an acceptable range. A Completion Time of 24 hours is permitted based upon the availability of temperature indication in the MCR. It is judged to be a sufficient amount of time allotted to correct the deficiency in the nonsafety ventilation system before shutting down.

C.1

If the MCR pressure boundary is damaged or otherwise degraded, action is required to restore the integrity of the pressure boundary and restore it to OPERABLE status within 24 hours. A Completion Time of 24 hours is permitted based upon operating experience. It is judged to be a sufficient amount of time allotted to correct the deficiency in the pressure boundary.

D.1 and D.2

In MODE 1, 2, 3, or 4 if Conditions A, B, or C cannot be restored to OPERABLE status within the required Completion Time, the plant must be placed in a MODE that minimizes accident risk. This is done by entering MODE 3 within 6 hours and MODE 5 within 36 hours.

E.1

During movement of irradiated fuel assemblies, if the Required Action A.1, B.1, or C.1 cannot be completed within the required Completion Time, the movement of fuel must be suspended. Performance of Required Action E.1 shall not preclude completion of actions to establish a safe condition.

F.1, F.2, and F.3

If the VES is inoperable in MODE 1, 2, 3, or 4, the VES may not be capable of performing the intended function, and the plant must be brought to MODE 4, where the probability and consequences of an event are minimized, and the VES must be

(continued)

RAI Number 630.049

BASES

ACTIONS

F.1, F.2, and F.3 (continued)

restored to OPERABLE status within 36 hours. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 4 within 12 hours.

G.1

During movement of irradiated fuel assemblies with the VES inoperable, the Required Action is to immediately suspend activities that present a potential for releasing radioactivity that might enter the MCR. This places the plant in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

The MCR air temperature is checked at a frequency of 24 hours to verify that the VBS is performing as required to maintain the initial condition temperature assumed in the safety analysis, and to ensure that the MCR temperature will not exceed the required conditions after loss of VBS cooling. The surveillance limit of 75°F is the initial heat sink temperature assumed in the VES thermal analysis. The 24 hour Frequency is acceptable based on the availability of temperature indication in the MCR.

SR 3.7.6.2

Verification every 24 hours that compressed air storage tanks are pressurized to [≥ 3400 psig] is sufficient to ensure that there will be an adequate supply of breathable air to maintain MCR habitability for a period of 72 hours. The Frequency of 24 hours is based on the availability of pressure indication in the MCR.

SR 3.7.6.3

VES air delivery isolation valves are required to be verified as OPERABLE. The Frequency required is in accordance with the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.6.4

VES air header isolation valves are required to be verified open at 31 day intervals. This SR is designed to ensure that the pathways for supplying breathable air to the MCR are available should loss of VBS occur. These valves should be closed only during required testing or maintenance of downstream components, or to preclude complete depressurization of the system should the VES isolation valves in the air delivery line open inadvertently or begin to leak.

SR 3.7.6.5

Verification that the air quality of the air storage tanks meets the requirements of Appendix C, Table C-1 of ASHRAE Standard 62 is required every 92 days. If air has not been added to the air storage tanks since the previous verification, verification may be accomplished by confirmation of the acceptability of the previous surveillance results along with examination of the documented record of air makeup. The purpose of ASHRAE Standard 62 states: "This standard specifies minimum ventilation rates and indoor air quality that will be acceptable to human occupants and are intended to minimize the potential for adverse health effects." Verification of the initial air quality (in combination with the other surveillances) ensures that breathable air is available for 11 MCR occupants for at least 72 hours.

SR 3.7.6.6

Verification that all VBS isolation valves are OPERABLE and will actuate upon demand is required every 24 months to ensure that the MCR can be isolated upon loss of VBS operation.

SR 3.7.6.7

Verification that each VES pressure relief isolation valve within the MCR pressure boundary is OPERABLE is required in accordance with the Inservice Testing Program. The SR is used in combination with SR 3.7.6.7 to ensure that adequate vent area is available to mitigate MCR overpressurization.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.6.8

Verification that the VES pressure relief damper is OPERABLE is required at 24 month intervals. The SR is used in combination with SR 3.7.6.6 to ensure that adequate vent area is available to mitigate MCR overpressurization.

SR 3.7.6.9

Verification of the OPERABILITY of the self-contained pressure regulating valve in each VES air delivery flow path is required in accordance with the Inservice Testing Program. This is done to ensure that a sufficient supply of air is provided as required, and that uncontrolled air flow into the MCR will not occur.

SR 3.7.6.10

Per Reference 1, a functional test is required to establish that one VES air delivery flow path, using the safety related compressed air storage tanks, pressurizes the MCR envelope to at least a positive 1/8 inch water gauge pressure relative to the surrounding spaces at the required air addition flow rate of 65 ± 5 scfm (Ref. 3). The test need not last 72 hours, only long enough to demonstrate the ability to achieve the required differential pressure. The MCR envelope leakage rate must be within the design capacity of the VES to pressurize the MCR for 72 hours. One air delivery flow path is tested on an alternating basis. The system performance test demonstrates that the MCR pressurization assumed in dose analysis is maintained.

REFERENCES

1. Section 6.4, "Main Control Room Habitability Systems."

2. Section 9.4.1, "Nuclear Island Non-Radioactive Ventilation System."

3. SECY-95-132, "Policy and Technical Issues Associated With The Regulatory Treatment of Non-Safety Systems (RTNSS) In Passive Plant Designs (SECY-94-084)," May 22, 1995.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.050

Question:

(Section 16.1, TS 3.7.10) Modify the Bases of TS 3.7.10 with additional justification for the 8-hour Completion time for the loss of SG isolation function conditions of Actions C and D. What, if any risk insights does the AP1000 PRA provide to support this Completion Time?

Westinghouse Response:

In the AP1000 design, failure to isolate the SG during a SGTR does not challenge core cooling unless there is also failure to depressurize the RCS and cool the core using feed-bleed cooling. The alternate SGTR mitigation using feed and bleed RCS cooling is very reliable because of the multiple redundant features available.

In the modeling of a SGTR in the AP1000 PRA, SG isolation is assumed to require manual operator actions although in many sequences (ie with PRHR HX success paths) it is not required. The operator action is identified as CID-MAN00. PRA Table 50-8 (Human Error Risk Importances – Risk Increase) shows the probability of failure of this operator action to be 1.84E-3/event and its RAW to be 6.34. As a result, the conditional CMF with SG isolation failed can be estimated to be the total CMF times the RAW for failure of this SG isolation or 2.41E-7/yr * 6.34 or 1.53E-6/yr.

One way of evaluating risk importance of the 8 hour AOT is contained in Regulatory Guide 1.177 (An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications). This Reg Guide suggests that the risk impact of an AOT change be determined by calculating the Incremental Conditional Core Damage Probability (ICCDP). The ICCDP is calculated by subtracting the baseline CDF from the conditional CDF with the subject equipment out of service and then multiplying by duration of a single AOT under consideration. This Reg Guide indicates that ICCDP values less than 5.0E-7 are considered to have a small quantitative impact on plant risk. For AP1000, the baseline at-power CDF is 2.41E-7 and the conditional CDF with the SG isolation capability failed 1.53E-6. The ICCDP for this 8 hour AOT is then:

	Alternate	AP1000	NRC Limit
SG Isolation AOT	1 hr	8 hr	
ICCDP	1.47E-10	1.18E-9	5.0E-7

With the AP1000 AOT of 8 hours, the ICCDP is more than two orders of magnitude less than the NRC criteria.

Design Control Document (DCD) Revision:

The following addition is made to TS Bases 3.7.10, C.1 and D.1:

C.1

With both valves in one or more PORV flow paths inoperable, action must be taken to isolate the flow path with a closed and deactivated valve. The valve must be deactivated to assure that the flow path will not be opened by a high pressure signal during the course of an SGTR event. This action places the flow path in a condition which assures the safety function is performed. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the SG isolation valves. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk in Reference 6.

D.1 and D.2

With two valves in one or more blowdown flow paths inoperable, action must be taken to isolate the flow path with a closed valve. This action places the flow path in a condition which assures the safety function is performed. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the SG isolation valves. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk in Reference 3.

The following additional reference is added to TS Bases 3.7.10:

REFERENCES

1. Section 10.3.2.2.3, "Power-Operated Atmospheric Relief Valves."
2. Section 10.4.8, "Steam Generator Blowdown System."
3. Regulatory Guide 1.177, 8/98, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications."

PRA Revision:

None

RAI Number 630.050

BASES

ACTIONS (continued)

B.1 and B.2

With one valve in one or more blowdown flow paths inoperable, action must be taken to isolate the flow path with a closed valve. This action places the flow path in a condition which assures the safety function is performed. A Completion Time of 72 hours to isolate the flow path is based on the availability of one OPERABLE blowdown flow path isolation valve which is fully capable of performing the required isolation function.

Since the blowdown isolation valve is not deactivated, periodic verification is required to assure that the flow path remains isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of status indications available in the control room, and other administrative controls, to ensure that the valve remains in the closed position.

C.1

With both valves in one or more PORV flow paths inoperable, action must be taken to isolate the flow path with a closed and deactivated valve. The valve must be deactivated to assure that the flow path will not be opened by a high pressure signal during the course of an SGTR event. This action places the flow path in a condition which assures the safety function is performed. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the SG isolation valves. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk in Reference 6.

D.1 and D.2

With two valves in one or more blowdown flow paths inoperable, action must be taken to isolate the flow path with a closed valve. This action places the flow path in a condition which assures the safety function is performed. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the SG isolation valves. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk in Reference 3.

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

Since the blowdown isolation valve is not deactivated, periodic verification is required to assure that the flow path remains isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of status indications available in the control room, and other administrative controls, to ensure that the valve remains in the closed position.

E.1 and E.2

If the SG isolation valves cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 with the RCS cooling provided by the RNS within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

The function of the SG isolation valves (PORV block valves (SGS-PL-V027A & B), PORVs (SGS-PL-V233A & B) and blowdown isolation valves (SGS-PL-V074A & B and SGS-PL-V075A & B)) is to isolate the steam generators in the event of SGTR, Loss of Feedwater or Feedwater Line Break. Stroking the valves closed demonstrates their capability to perform the isolation function. The Frequency for this SR is in accordance with the Inservice Testing Program.

REFERENCES

1. Section 10.3.2.2.3, "Power-Operated Atmospheric Relief Valves."

2. Section 10.4.8, "Steam Generator Blowdown System."

3. Regulatory Guide 1.177, 8/98, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications."

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.051

Question:

(Section 16.1, TS Section 3.9) Acceptance of TS Section 3.9 requires resolution of RAI 410.009. Please address this issue and revise the DCD/TSs accordingly.

NOTE: AP1000 RAI 410.009 was issued on September 30, 2002 (ADAMS Accession No. ML022680338).

Westinghouse Response:

There are no changes to HVAC Technical Specifications (TS) as a result of the Westinghouse review of the AP1000 HVAC codes and standards per RAI 410.009.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.052

Question:

(Section 16.1, TS Section 3.9) STS Section 3.9 does not contain specifications for a decay time limit and a spent fuel pool fuel handling crane load limit. Nevertheless, such specifications must be included in the AP1000 TSs unless a suitable justification is provided, because these "operating restrictions" are directly related to assumptions of the design-basis fuel handling accident in the spent fuel storage pool, and satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). Provide design-specific justifications for omitting these LCOs from the AP1000 TSs.

Westinghouse Response:

The AP1000 technical specifications do not include a specification on decay time limit because the doses resulting from a fuel handling accident in the spent fuel handling area are not sensitive to the decay time assumed in the analysis. In the AP1000, a time of 100 hours after shutdown was assumed for the time of the accident to occur. The resulting doses are less than 25% of the dose guideline identified in 10 CFR Part 50.34. A sensitivity study was performed in which the fuel handling accident was assumed to occur just 24 hours after shutdown, even though this is an unrealistically short time for fuel movement. The resulting doses are still less than the guideline in 10 CFR Part 50.34. Therefore, the results of the fuel handling accident analysis are not sensitive to the decay time assumed for the dropped assembly, and thus a technical specification is not necessary to ensure safe fuel handling operations.

The AP1000 technical specifications do not include a specification on spent fuel pool fuel handling crane load limit because the AP1000 spent fuel handling machine is limited to lifting only one fuel assembly at a time. The fuel handling machine, which is the only machine to handle spent fuel in the spent fuel storage area, lifts each fuel assembly into a mast before it can be moved. The design of the machine allows only one assembly in the mast. Therefore, by design, the maximum load the machine will lift is limited to one fuel assembly plus handling tools. Since the spent fuel handling machine is limited to lifting one fuel assembly, and the fuel handling accident assumes one dropped fuel assembly, a technical specification on the fuel handling machine load limit is not necessary to ensure safe fuel handling operations.

In previous NRC reviews of other plant designs, Refueling Operations Technical Specifications without a decay time limit and a spent fuel pool fuel handling crane load limit were found acceptable. Both the AP600 and the System 80+ received design certification without including these two technical specifications.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.053

Question:

(Section 16.1, TS Section 5.5) Explain the basis for not adopting STS 5.5.3, Post Accident Sampling.

Westinghouse Response:

Technical Specifications related to Post Accident Sampling have not been included in the AP1000, consistent with the elimination of post accident sampling capability in the design of the primary sampling system as given in DCD section 9.3.3.5. Post accident sampling capability was not included in the AP1000 primary sampling system design in accordance with Reference 5 of DCD Section 9.3 (Safety Evaluation by the Office of Nuclear Regulation Related to WCAP-14986, "Westinghouse Owners Group Post Accident Sampling System Requirements", Westinghouse Owners Group Project No. 694, June 14, 2000).

In this reference the NRC agreed with the Westinghouse Owners Group report (WCAP-14986) that the current post accident sampling system requirements can be eliminated. This decision is based on the current understanding of core damage accidents, the current guidance for bringing the plant to a safe stable state following a core damage accident, and the basis for emergency planning decision making.

Deletion of post accident sampling technical specifications is consistent with STS 5.5.3 which includes a note that the post accident sampling program may be eliminated based on implementation of WCAP-14986 and the associated NRC safety evaluation.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.039

Question:

An important objective of the AP600 PRA was to provide input to the design certification process regarding the need for regulatory oversight of certain non-safety-related systems. The process used to identify SSCs for regulatory oversight as well as the type and level of such oversight was called Regulatory Treatment of Non-Safety-Related Systems (RTNSS) in the AP600 certification. The end results of this process were the "availability controls" documented in Section 16.3 of the DCD. Please provide similar information for AP1000. This information should account for uncertainties in the AP1000 PRA so that it can be used by the staff to make similar conclusions, about the need for non-safety-system oversight, as were made for the AP600 design (e.g., as documented in the AP600 FSER Chapter 19.1.7 "PRA input to the RTNSS Process.)"

Westinghouse Response:

WCAP-15985 "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process" provides the basis for the implementation of the RTNSS policy statement. The AP1000 implementation of the RTNSS policy statement is consistent with the approach that was taken for the AP600. Note that PRA sensitivity studies with nonsafety systems failed were used to evaluate the importance of nonsafety features in PRA accident mitigation, instead of the focused PRA sensitivity study. The AP1000 PRA, revision 0, includes the CMF sensitivity study; the LRF sensitivity study was performed recently based on the response to RAI 720.057 and will be added to the PRA in revision 1.

Use of these sensitivity studies increases the need for passive safety feature actuation signals since non-safety AC power is assumed to be available after each accident except for loss of offsite power. As a result, some non-safety manual Diverse Actuation System (DAS) controls are required to meet the licensing PRA safety goals. These manual DAS controls are captured as RTNSS important and additional regulatory oversight is provided. Since these manual DAS controls meet the technical specification screening criteria for PRA importance, a Technical Specification is added on these manual DAS controls. The additional Technical Specification is included in the revised AP1000 Technical Specifications that are submitted as part of the response to RAI 630.001.

These DAS manual controls were the only additional non-safety featured captured in the AP1000 RTNSS evaluation. The list of non-safety features that have short-term availability controls is the same for AP1000 and AP600.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

See the attached Technical Specification for DAS that is included in the revised AP1000 Technical Specifications.

PRA Revision:

None

RAI Number 720.039

3.3 INSTRUMENTATION

3.3.5 Diverse Actuation System (DAS) Manual Controls

LCO 3.3.5 The DAS manual controls for each function in Table 3.3.5-1 shall be operable.

APPLICABILITY: According to Table 3.3.5-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more manual DAS controls inoperable.	A.1 Restore DAS manual controls to OPERABLE status.	30 days
B. Completion Time of Required Action A not met for inoperable DAS manual reactor trip control.	B.1 Perform SR 3.3.1.5. AND B.2 Restore all controls to OPERABLE status.	Once per 31 days on a STAGGERED TEST BASIS Prior to entering MODE 2 following next MODE 5 entry
C. Completion Time of Required Action A not met for inoperable DAS manual actuation control other than reactor trip.	C.1 Perform SR 3.3.2.3. AND C.2 Restore all controls to OPERABLE status.	Once per 31 days on a STAGGERED TEST BASIS Prior to entering MODE 2 following next MODE 5 entry

(continued)

RAI Number 720.039

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Completion Time of Required Action B not met. OR Completion Time of Required Action C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	----------NOTE ---------- Verify of setpoint not required. ---------- Perform TRIP ACTUATION DEVICE OPERATIONAL TEST (TADOT).	 24 months

RAI Number 720.039-4

RAI Number 720.039

B 3.3 INSTRUMENTATION

B 3.3.5 Diverse Actuation System (DAS) Manual Controls

BASES

BACKGROUND

The Diverse Actuation System (DAS) manual controls provide backup controls in case of common mode failure of the Protection and Safety Monitoring System (PMS) automatic and manual actuations evaluated in the AP1000 PRA. These controls are not credited for mitigating accidents in the DCD Chapter 15 analyses.

The specific DAS controls were selected based on PRA risk importance as discussed in Reference 1. As noted in Reference 1, electrical power for these controls and instrument indications need not be covered by Technical Specifications. The rational is that these controls use the same nonsafety-related power supply used by the plant control system. This power is required to be available to support normal operation of the plant. With offsite power available, there are several sources to provide this power including AC power to non-Class 1E battery chargers, AC power to rectifiers, and non-Class 1E batteries. As a result, with offsite power available it is very likely that power will be available for these DAS controls. If offsite power is not available, then there is still the likelihood that the non-1E batteries or the non-1E diesel generators will be available. Even if these sources are unavailable, the desired actions will occur without operator action for the more probable events. The rods will insert automatically on loss of offsite power. The passive residual heat removal heat exchanger (PRHR HX), core makeup tanks (CMT), passive containment cooling system (PCS), and containment isolation features are initiated by operation of fail-safe, air-operated valves. If all offsite and onsite AC power is lost, the instrument air system will depressurize by the time these functions are needed in the 1-hour time frame.

Instrument readouts are expected to be available even in case of complete failure of the PMS due to common cause failure. These instruments include both DAS and PLS instruments. As discussed above, it is expected that AC power will be available to power the instruments. Even if the operators have no instrument indications, they are

(continued)

RAI Number 720.039-5

RAI Number 720.039

BASES

BACKGROUND (continued)

expected to actuate the controls most likely to be needed (PRHR HX, CMT, PCS, and containment isolation). If all AC power fails, then the rods will drop and the air-operated valves will go to their fail-safe positions.

The DAS uses equipment from sensor output to the final actuated device that is diverse from the PMS to automatically initiate a reactor trip, or to manually actuate the identified safety-related equipment. DCD Section 7.7.1.11 (Ref. 2) provides a description of the DAS.

APPLICABLE SAFETY ANALYSES

The DAS manual controls are required to provide a diverse capability to manually trip the reactor and actuate the specified safety-related equipment, based on risk importance in the AP1000 PRA.

The DAS manual controls are not credited for mitigating accidents in the DCD Chapter 15 safety analyses.

The AP1000 PRA, Appendix A, provides additional information, including the thermal and hydraulic analyses of success sequences used in the PRA.

The DAS manual controls satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The DAS LCO provides the requirements for the OPERABILITY of the DAS manual trip and actuation controls necessary to place the reactor in a shutdown condition and to remove decay heat in the event that the PMS automatic actuation and manual controls are inoperable.

APPLICABILITY

The DAS manual controls are required to be OPERABLE in the MODES specified in Table 3.3.5-1.

The manual DAS reactor trip control is required to be operable in MODES 1 and 2 to mitigate the effects of an ATWS event occurring during power operation.

The other manual DAS actuation controls are required to be available in the plant MODES specified, based on the need

(continued)

RAI Number 720.039-6

RAI Number 720.039

BASES

APPLICABILITY (continued)

for operator action to actuate the specified components during events that may occur in these various plant conditions, as identified in the AP1000 PRA.

ACTIONS

A.1

Condition A applies when one or more DAS manual controls are inoperable.

The Required Action A.1 to restore the inoperable DAS manual control(s) to OPERABLE status within 30 days is reasonable because the DAS is a separate and diverse non-safety backup system for the manual reactor trip and manual safety-related equipment actuation controls. The 30-day Completion Time allows sufficient time to repair an inoperable manual DAS control but ensures the control is repaired to provide backup protection.

B.1 and B.2

Condition B applies when Required Action A cannot be completed for the DAS manual reactor trip control within the required completion time of 30 days.

Required Action B.1 requires SR 3.3.1.5, "Perform TADOT" for the reactor trip breakers, is to be performed once per 31 days, instead on once every 92 days. The predominant failure requiring the DAS manual reactor trip control is common mode failure of the reactor trip breakers. This change in surveillance frequency for testing the reactor trip breakers increases the likelihood that a common mode failure of the reactor trip breakers would be detected while the DAS manual reactor trip control is inoperable. This reduces the likelihood that a diverse manual reactor trip is required. It is not required to perform a TADOT for the manual actuation control. The manual reactor trip control is very simple, highly reliable, and does not use software in the circuitry.

Action B.2 requires that the inoperable DAS manual reactor trip control be restored to OPERABLE status prior to entering MODE 2 following any plant shutdown to MODE 5 while the control is inoperable. This ACTION is provided to ensure that all DAS manual controls are restored to OPERABLE status following the next plant shutdown.

(continued)

RAI Number 720.039-7

RAI Number 720.039

BASES

ACTIONS (continued)

C.1 and C.2

Condition C applies when Required Action A cannot be completed for any DAS manual actuation control (other than reactor trip) within the required completion time of 30 days.

Required Action C.1 requires SR 3.3.2.2, "Perform ACTUATION LOGIC TEST," to be performed once per 31 days, instead on once every 92 days. The predominant failure requiring the DAS manual reactor trip control is common mode failure of the PMS actuation logic software or hardware. This change in surveillance frequency for actuation logic testing increases the likelihood that a common mode failure of the PMS actuation logic from either cause would be detected while any DAS manual actuation control is inoperable. This reduces the likelihood that a diverse component actuation is required. It is not required to perform a TADOT for the manual actuation control device since the manual actuation control devices are very simple and highly reliable.

Action C.2 requires that the inoperable DAS manual actuation control(s) be restored to OPERABLE status prior to entering MODE 2 following any plant shutdown to MODE 5 while the control is inoperable. This ACTION is provided to ensure that all DAS manual controls are restored to OPERABLE status following the next plant shutdown.

D.1 and D.2

Condition D is entered if the Required Action associated with Condition B or C is not met within the required Completion Time.

Required Actions D.1 and D.2 ensure that the plant is placed in a condition where the probability and consequences of an event are minimized. The allowed Completion Times are reasonable based on plant operating experience, for reaching the required plant conditions from full power conditions in an orderly manner, without challenging plant systems.

(continued)

RAI Number 720.039

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1

SR 3.3.5.1 is the performance of a TADOT of the DAS manual trip and actuation controls for the specified safety-related equipment. This TADOT is performed every 24 months.

The Frequency is based on the known reliability of the DAS functions and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of the setpoints from the TADOT. The functions have no setpoints associated with them.

REFERENCES

1. WCAP-15985, "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," November 2002.

2. DCD, Section 7.7.1.11.